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ABSTRACT

This technical specialty course is one of thirty-five courses designed for use by two-year postsecondary institutions in five nuclear technician curriculum areas: (1) radiation protection technician, (2) nuclear instrumentation and control technician, (3) nuclear materials processing technician, (4) nuclear quality-assurance/quality-control technician, and (5) nuclear reactor (plant) operator trainee. This course, Radiation Protection I, contains six modules: Radiation Exposure and Absorbed Dose Values; Biological Effects of Radiation; Radiation Protection Standards; Internal Dose Calculations; Personnel Monitoring Devices, Ionization Types; and Personnel Monitoring Devices, Chemical Types. Each module follows a typical format including the following sections: introduction, module prerequisites, objectives, subject matter, exercises, materials, procedures, data tables, reference materials, and tests. (A master index and suggested program planning guides are also available--see CE 026 856 and CE 026 639-643.) (LRA)

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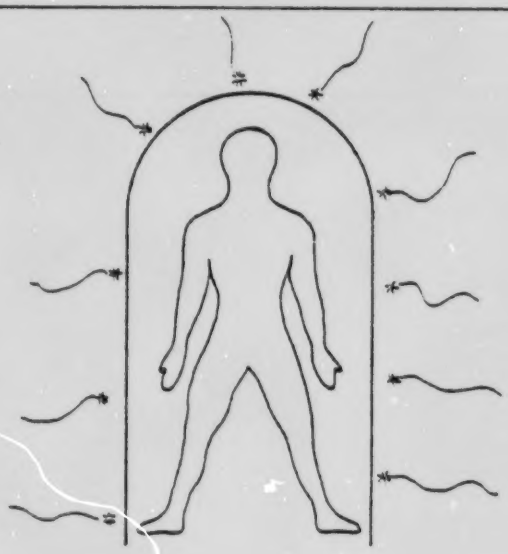
NUCLEAR TECHNOLOGY

Series

MODULE 2-1 RADIATION EXPOSURE AND ABSORBED DOSE VALUES



RADIATION PROTECTION I

COURSE 2

CE 026 645

U.S. DEPARTMENT OF HEALTH
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COURSE MODULES

COURSE 2 RADIATION PROTECTION I

MODULE 2-1 Radiation Exposure and Absorbed Dose Values

MODULE 2-2 Biological Effects of Radiation

MODULE 2-3 Radiation Protection Standards

MODULE 2-4 Internal Dose Calculations

MODULE 2-5 Personnel Monitoring Devices, Ionization Types

MODULE 2-6 Personnel Monitoring Devices, Chemical Types

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INTRODUCTION

X-ray, the first type of radiation to be given a unit of exposure, called the Roentgen (abbreviated R), was introduced at the Radiological Congress in 1928. Since then, R has been the most widely used unit of x-ray exposure.

> The Roentgen shall be the quantity of x or gamma radiation such that the associated corpuscular emission per 0.001293 grams of air produces in air, ions carrying one electrostatic unit of electrical charge of either sign.

This definition has been superseded by one stated in terms of MKSA* units. "One Roentgen equals 2.58×10^{-4} coulombs per kilogram of air." The two definitions are equivalent.

Unfortunately, though useful, measurement of exposure has been difficult to compare from one situation to another, and various units came and went until, in 1962, the International Commission on Radiological Units (ICRU) explicitly defined the various radiological quantities and units in its Report 10A, Radiation Quantities and Units.

As both measurements and concepts became more sophisticated, a much greater precision was needed in many quantities dealing with radiation. In Report 11 (September, 1968), the ICRU made recommendations concerning new definitions:

Directly ionizing particles.

Indirectly ionizing particles.

Absorbed doses.

Exposures.

Radiation has been perhaps the most thoroughly investigated causative agent associated with disease. Much more

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* MKSA - The system of units in which distance is measured in meters, mass in kilograms, time in seconds and electric charge in coulombs.

knowledge is needed, of course, but more is known about the mechanism of radiation damage on the molecular, cellular and system levels than almost any other deleterious agent affecting man. Because of this vast amount of knowledge, radiation protectionists (health physicists) are able to recommend levels of exposure generally accepted as safe for those persons employed in the nuclear industry. As a result, nuclear energy may be developed and employed in relative safety.

MODULE PREREQUISITES

The student should have a knowledge of basic algebra, laboratory safety, and basic radiation, including the decay law and the interaction of radiation with matter, as presented in Course 1, "Radiation Physics."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Define in terms presented in this module:
   - Roentgen.
   - Rad.
   - Rem.
   - Quality factor.

2. Convert from rads to rems.
3. Explain why quality factor is different for different types of radiation.
4. a. Given a radionuclide, determine its Γ number and calculate the dose rate at one meter from a one-curie point source.
   b. Calculate dose rate from the same source at different distances.

THE ROENTGEN

A fascinating area of science deals with the effects of radiation on humans. As indicated earlier, man has sought answers to the consequences of using radiation for many years. During the early days, there were no precise units of radiation that could be used either by the radiation protectionist or the radiation therapist. The radiation pioneers literally estimated doses by skin color. An early dose was referred to as a "Skin Erythema Unit" - the amount of x-ray radiation needed to turn the recipient's skin a slight red color.

Since then, "radiation protection guides" have been formulated by many organizations who have set standards:

International X-Ray and Radiation Protection Commission.

International Commission on Radiological Protection (1928).

International Atomic Energy Agency (1956).

International Labor Organization (1919).

International Commission on Radiological Units and Measurements (1925).

National Committee on Radiation Protection and Measurements (1929).

Federal Radiation Council (1959).

An important milestone, ICRU's 1962 report, explicitly defined units then in use. The following discussion defines these terms and explains why they were adopted.

Exposure refers to radiation that exists with nothing but air present. The numbers used to specify exposure indicate how much energy is being deposited in a kilogram of

standard air by the radiation. The unit of exposure is the Roentgen (R) - "That quantity of x or gamma radiation that produces 2.58×10^{-4} coulombs per kilogram of air." This definition may also be expressed as, "That quantity of x or gamma radiation that produces 1 statcoulomb per cubic centimeter of air."

An "exposure" indicates that radiation is absorbed in air. It also indicates that it must be x or gamma (γ), and applies to no other radiation.

This operational definition of the Roentgen may be easily converted into more fundamental units of energy absorbed per unit mass of air by applying (1) the charge on a single ion, which is 4.8×10^{-10} statcoulombs (sC) and (2) the average energy dissipated in production of a single ion pair in air, 34 electronvolts (eV):

$$1\ R = \left(\frac{1\ \text{statcoulomb}}{\text{cm}^3\ \text{air}}\right)\left(\frac{1\ \text{cm}^3\ \text{air}}{0.001293\ \text{g}}\right)\left(\frac{1}{4.8 \times 10^{-10}}\ \frac{\text{ion}}{\text{statcoulomb}}\right)\left(\frac{34\ \text{eV}}{\text{ion}}\right)$$

$$\left(1.6 \times 10^{-12}\ \frac{\text{ergs}}{\text{eV}}\right)$$

$1\ R = 87.7\ \text{ergs/g air.}$ Equation 2-1-1
(Equation = Eq.)

It should be noted that R is a measure of exposure, independent of time period of exposure. The intensity of a radiation field is usually given as an exposure rate, such as Roentgens per hour (R/hr) or milliroentgens per hr (mR/hr). (A milliroentgen equals 0.001 Roentgen.) Total exposure is the product of exposure rate and exposure time.

As the Roentgen is a measure of exposure to radiation, the curie is a measure of radiation source strength. At one time, a gram of radium was taken to be of one-curie source strength. Now, the unit of radioactivity, the curie (Ci), is defined as 3.7×10^{10} disintegrations per second with no reference to radiation energy.

Simply knowing the source strength does not specify exposure rate. The units must be related to each other by a "specific gamma ray constant" (Γ). Table 2-1-1 gives the Γ value in R-cm^2/hr-mCi, which - when divided by 10 - gives R/hr at one meter from a one-curie source of the given radionuclide.

Notice that Table 2-1-1 shows units of R per hour, not R. The reason for this is that a Roentgen is a unit of radiation intensity, with no time units involved. It takes a curie of radium-226 one hour to expose one cubic centimeter of air to 0.825 R. This comparison is analogous to the difference between "work" and "power": the Roentgen defines how much energy is being deposited, as does work, and Roentgen per unit of time indicates the rate of energy deposition, as does power.

The Roentgen is a unit of exposure, and the R per hour is a measure of exposure rate. Radiation protection personnel post warning signs in areas or on equipment indicating radiation level on or near the area. This posting satisfies a legal requirement established by the Federal government. In contrast to recommendations that can be used by various technical societies, the government can and does issue rules that must be followed in the nuclear industry. Such rules are published in a Code of Federal Regulations (CFR). In this example, Title 10 of the CFR, Part 20, Standards for Protection Against Radiation, states that

TABLE 2-1-1

GAMMA RADIATION LEVELS FOR ONE CURIE OF SOME RADIONUCLIDES*

NUCLIDE	Γ†	NUCLIDES	Γ†	NUCLIDES	Γ†
Actinium - 227	~2.2	Gold - 198	2.3	Potassium - 43	5.6
Antimony - 122	2.4	Gold - 199	~0.9	Radium -226	8.25
Antimony - 124	9.8	Hafnium - 175	~2.1	Radium - 228	~8.1
Antimony - 125	~2.7	Hafnium - 181	~3.1	Rhenium - 186	~0.2
Arsenic - 72	10.1	Indium - 114M	0.2	Rubidium - 86	0.5
Arsenic - 74	4.4	Iodine - 124	7.2	Ruthenium - 106	1.7
Arsenic - 76	2.4	Iodine - 125	~0.7	Scandium - 46	10.9
Barium - 131	~3.0	Iodine - 126	2.5	Scandium - 47	0.56
Barium - 133	~2.4	Iodine - 130	12.2	Selenium - 75	2.0
Barium - 140	12.4	Iodine - 131	2.2	Silver - 110m	14.2
Beryllium - 7	~0.3	Iodine - 132	11.8	Silver - 111	~0.2
Bromine - 82	14.6	Iridium - 192	4.8	Sodium - 22	12.0
Cadmium - 115m	~0.2	Iridium - 194	1.5	Sodium - 24	18.2
Calcium-47	5.7	Iron - 59	6.4	Strontium - 85	3.0
Carbon - 11z	5.9	Krypton-85	~0.04	Tantalum-182	6.8
Cerium - 141	0.25	Lanthanum-140	11.3	Tellurium-121T	3.3
Cerium - 144	~0.4	Lutecium-177	0.09	Tellurium-132	2.2
Cesium-134	9.7	Magnesium -28	15.7	Thulium-170	0.025
Cesium-137	3.3	Manganese-52	19.6	Tin-113	~1.7
Chlorine -38Z	8.8	Manganese-54	4.7	Tungsten-185	0.5
Chromium-51	0.16	Manganese-56	8.3	Tungsten-187	3.0
Cobalt-56	17.6	Mercury-197	~0.4	Uranium-234	~0.1
Cobalt-57	0.9	Mercury 203	1.3	Vanadium-48	15.6
Cobalt-58	5.3	Molybdenum-99	~1.8	Xenon-133	0.1
Cobalt-60	13.2	Neodymium-147	3.8	Ytterbium-175	3.4
Copper-64	1.2	Nickel-65	~3.1	Yttrium-88	14.1
Europium-152	5.8	Niobium-95	4.2	Yttrium-91	0.01
Europium-154	~4.2	Osmium-191	~0.6	Zinc-65	2.7
Europium-155	~0.3	Palladium-109	0.02	Zirconium-95	4.2
Gallium-67	~1.1	Platinum-197	~0.8		
Gallium-72	12.6	Potassium-42	1.4		

* Jaeger, R. G., et al., Engineering Compendium on Radiation Shielding, Vol. 1. New York: Springer-Verlag, 1968, pp. 11-20.

† Γ = R-m²/hr-mCi or Γ/10 = R/hr at 1 m fr

z A Manual of Radioactivity Procedures, National Bureau of Standards Handbook No. 80, Washington, D.C.: 8.20. 22 Docs., U.S. Government Printing Office, Nov. 1961), Appendix A, pp 117-142.

areas with radiation levels greater than 100 mR per hour must be posted with a "High-Radiation Area" sign. Entry of personnel into such an area is restricted to a much greater degree than entry into areas of intensity less than 100 mR per hour.

It has become common in high-intensity areas to post intensity in mR units, rather than R, although a requirement still exists for a High-Radiation Area sign. For example, the sign for some high-radiation areas might be posted as, say, 1,500 mR per hour, rather than 1.5 R per hour, even though the latter would be equally valid. The important thing to remember is that the area is severely limited as far as access is concerned and, under Federal regulations, the dose rate must be reduced to below 100 mR per hour or have control devices installed to keep all unauthorized individuals out of the area.

Absorbed dose refers to the amount of energy absorbed by a material when exposed to radiation. It was shown that an exposure of one Roentgen leads to the absorption of approximately 88 ergs of energy in 1 g of air, but absorption in air is usually of little interest. In most situations, it is of more interest to know what the energy deposition is in a specific material. For example, radiation damage in human tissue results from the absorption of energy from radiation and the degree of damage is proportional to the amount of energy absorbed in the tissue. For this reason, the basic unit of radiation dose is expressed in terms of "absorbed energy per unit mass of tissue."

An absorbed dose does not specify tissue but, because radiation protectionists are primarily interested in protection of people, they are interested in a tissue unit rather than, say, one for the cement used as a shield.

THE RAD

The rad is a unit of energy absorption. Rate of energy absorption is the rad per unit time (or millirad per unit time). A basic part of the radiation protection program consists of determining or knowing anticipated dose rate. Such information allows calculation of the amount of time an individual can spend under these conditions and still maintain an absorbed dose within acceptable limits.

Obviously, it is more important to know this dose rate before an exposure begins - once radiation has been absorbed, the tissue cells experience some deleterious effect. Control of exposure, in most instances, is interpreted to mean minimizing it.

Exposure to radiation is often minimized by simply avoiding it. If it cannot be avoided, then the time spent in a radiation field must be minimized and, finally, if neither alternative is possible, temporary or permanent shielding should be installed. (This time/distance shielding concept will be used over and over.)

THE ROENTGEN - RAD RELATIONSHIP

As indicated earlier, a Roentgen is the unit of measure for exposure and the rad is the unit of measure for absorbed dose. Because the energy absorbed (absorbed dose) is the quantity that determines radiation effects on human tissue, it is appropriate to establish the relationship between the Roentgen and the rad, between exposure and dose.

When radiation, particularly x-ray or gamma radiation, interacts with matter, the degree of interaction is a function of the electron density of the material; therefore, consider tissue to have a density of 1 g/cm^3, consisting of 5.98 x 10^{22} hydrogen atoms, 2.76 x 10^{22} oxygen atoms, 1.72 x 10^{21} nitrogen atoms and 6.02 x 10^{21} carbon atoms per gram. By considering the atomic numbers of the various elements, there will be 3.28 x 10^{23} electrons per gram. For air at 1.293 x 10^{-3} grams per cubic centimeter there are 3.01 x 10^{23} electrons per gram. The energy absorption in ergs per gram of tissue corresponding to 1 R in air is therefore 3.28 ÷ 3.01 x 87.7 ≈ 95 ergs per gram tissue. This air dose of 95 ergs per gram tissue from a 1 R exposure is very close to 100 ergs per gram. A rad is defined as 100 ergs/g absorbed dose. Figure 2-1-1 gives some idea of how this conversion factor varies with different biological materials as a function of γ-ray energy. Remember that the Roentgen is exposure, whereas the rad is the measure of the energy absorbed in any material. Thus, the figure indicates how much energy is being absorbed for each Roentgen of exposure at different values of γ energy.

The abscissa - energy scale - (Figure 2-1-1) is drawn on a log scale. Note that, in the range from 0.2 to 5 MeV, the curves for fat and muscle are fairly flat and only slightly higher than the curve for bone. For gamma radiation in this energy range, one Roentgen of exposure results in about 100 ergs per gram absorbed, or 1 rad of absorbed dose.

For γ-ray energies outside the range of from 0.2 to 2 MeV, the energy deposition will be less in muscle and fat; so, if it were still assumed that one Roentgen produces one rad, this would be a conservative estimate. In the case of bone, it would not be conservative if the energy

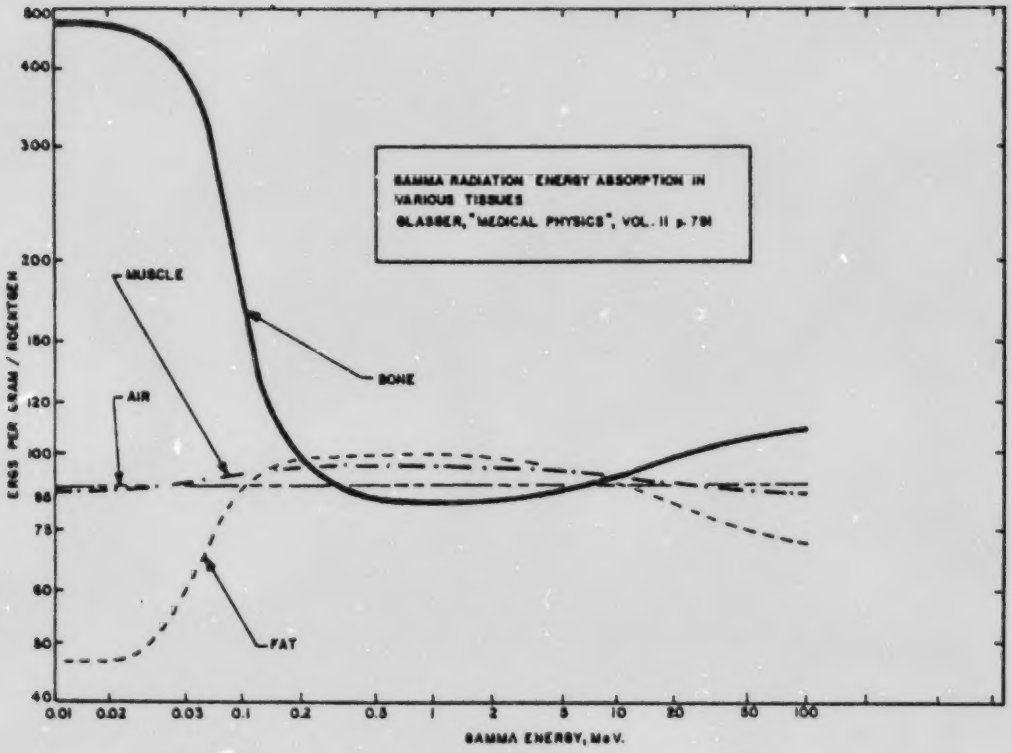

Figure 2-1-1. Gamma radiation.

of the γ rays were over 20 MeV or under 0.2 MeV; however, 20-MeV γ rays rarely occur in disintegration processes (check the Chart of the Nuclides or any other listing of γ-ray energies). For a considerable band of gamma energies, the rule of thumb that one Roentgen equals one rad in the body is acceptable.

But what about other types of radiation? It makes little difference what the radiation is, because the rad is defined as 100 ergs per gram absorbed dose.

THE REM

The biological effect describes the effect of radiation upon the tissue of the body. The unit for the biological dose is the dose equivalent (DE), represented in the rem. This unit is used for radiation safety purposes, as well as administrative purposes.

The dose equivalent expressed in rems considers the quality factor of radiation (discussed later), as well as the absorbed dose and other factors, such as nonuniform distribution, that may influence the biologic effect of a given absorbed dose.

The dose equivalent is determined as follows:

$$DE_{rems} = rads \times QF \times DF \qquad \text{Eq. 2-1-2}$$

where

QF = quality factor.

DF = distribution factor.

In the basic definition of a Roentgen, it was stated that the Roentgen pertained only to x and γ radiation. Other types of radiation have differing cellular effects. Why? Because of their specific ionizing characteristics and the manner in which they interact with matter. An x-ray, for example, may cause a dozen ion pairs to be formed within a cell of the body, and the cell may not be adversely affected by this. But an alpha particle can cause many, many more ion pairs to be formed within a cell.

When biologic damage is mentioned, it generally means

that an observable sign or symptom is present. If radiation is absorbed, some damage takes place. But, if the damage is minimal, no observable indication appears. (A later module explores the total spectrum of biological effects. For now, however, remember that biological damage is very difficult to explain and, in fact, has no universally acceptable definition.)

When dealing with man, there are too many variables, and the extremes of effects run from "effects almost universally detrimental" to an opinion that "a little radiation is good for everyone."

Understanding of effects of radiation has improved markedly in recent years. Also, it is possible, on the basis of data on experimental animals and from limited observations in man, to arrive at exposure criteria that entail minimal risk to an individual (and descendants). An assumption can be made that present-day guidelines seem adequate to protect the individual, as well as society.

QUALITY FACTOR

For an example of deposition of energy for different radiations, consider the neutron. A neutron interacts (collides) with a water molecule in a living cell, causing a proton to recoil, and slowing down the neutron. The high-speed recoil proton is almost as highly ionizing as an alpha (α) particle, so a high-energy (fast) neutron can cause a higher specific ionization within a cell than can a low-energy (slow) neutron.

If the body absorbs one rad of each different type of radiation, though the same amount of energy is deposited, the biological effect depends, in large measure, on the spatial distribution of that energy. Extensive studies have shown that an amount of heavy ion radiation sufficient to deposit 100 ergs per gram of energy in the body can cause twenty times as much cell damage as will one rad of x or γ radiation. This "factor of twenty" is called the quality factor (QF). This factor is given in Table 2-1-2 for various types of radiation.

TABLE 2-1-2
GENERAL RADIATION QUALITY FACTORS

Type of Radiation	Typical QF	
α	10	
β	1	
β^- < 0.03 MeV	1.7	
x and γ	1	
Slow neutrons	3	See Table
Fast neutrons	10	2-1-3.
Heavy ions	20	

The equation relating rad to rem now becomes:

$$\text{rad} \times QF = DE_{rem} \text{ or } QF = DE_{rem} \text{ per rad.}$$

In radiation protection, it is necessary to provide a factor that denotes modification of effectiveness of a given absorbed dose by linear energy transfer (LET)*. Unlike relative biological effectiveness (RBE), which is always experimentally determined, this factor is assigned on the basis of a number of considerations. A full discussion of these considerations is beyond the scope of this module. It is sufficient to recognize that the distribution of the absorbed energy influences the effects.

Because of the wide energy spectrum of neutrons, Table 2-1-3 gives the relationship between neutron energy and QF, and neutron energy and neutrons per centimeter^2 per second, to give an absorbed dose of 100 millirem per 40 hours.

TABLE 2-1-3
RADIATION QUALITY FACTOR FOR NEUTRONS

$^{1}_{0}n$ Energy (MeV)	QF	100 mrem/40 hr n/cm^2 per sec
Thermal	3	670
0.0001	2	500
0.005	2.5	570
0.02	5	280 (ther)
0.1	8	80
0.5	10	30
1.0	10.5	18
2.5	8	20
5.0	7	18
7.5	7	17
10.0	6.5	17

* LET - linear energy transfer is the amount of energy deposited by the ionizing radiation per unit path length traveled in the absorbing medium.

The term relative biological effectiveness (RBE) was the term in popular use before QF existed. It indicated the ratio of the amount of energy required to produce a known effect to the energy of 200 keV x-rays to produce the same effect. Now, however, RBE is restricted in application to radiation biology. In this discipline, experimenters use exact species of organisms under very rigid sets of circumstances and conditions. This application produces an exact biological effect under experimental conditions. In radiation protection, a conservative upper limit of the RBE is used.

Consider now beta radiation. In the original definition of a Roentgen, it was pointed out that gamma radiation produced "corpuscular emission that is in some respects similar to beta particles." These secondary electrons act the same as low-energy beta particles. Thus, if a gamma ray has to affect the body cells by first resulting in the emission of electrons, then a beta particle should have the same effect on the body as a gamma ray. It does, which is why both beta and gamma radiation have the same quality factor (Table 2-1-2).

For beta radiation, a rad is a rem. For radiation other than gamma and x-rays, it is improper to use the terms R, or mR (some people mistakenly speak of mR per hour from beta emitters). Refer to beta absorbed dose as rads. As with rad and R, it is proper to speak of the dose rate when meaning the dose equivalent in rem per hour or mR per hour.

SUMMARY

By the use of a quality factor and such other factors as may be necessary, it is possible to equate various energies of different radiations to a biological effect. It is also possible to equate air dose to dose in tissue by a rad-Roentgen relationship, where the rad is equal to 100 ergs per gram of absorbed energy in any material. One R exposure, approximately 87.7 ergs absorbed per gram of air, is equivalent to approximately 95 ergs absorbed per gram of tissue.

Relative biological effectiveness (RBE), the forerunner of QF, has been reserved for the science of radiation biology, while QF applies to radiation protection. The concept of time, distance and shielding is used to limit the amount of radiation absorbed by radiation workers.

EXERCISES

None.

MATERIALS

1 encapsulated gamma source about which the following is known:

Source activity.

Date on which the activity was calibrated.

Source material.

1 ion chamber type ratemeter.

PROCEDURES

1. Familiarize yourself with ratemeter operation. This instrument will be used to determine accuracy of calculations you perform.
2. Observe ratemeter; enter value of reading on scale (lower two ranges of meter) on Data Table 2-1-2.
3. Fill in source data on the data sheet.
4. By using "decay law," calculate source activity today.
5. From Table 2-1-1, value for source material is _____. This value is R per hour at one meter. Use inverse-square law to determine distance at which desired reading will occur. (Inverse-square law:

$$I_1 d_1^2 = I_2 d_2^2$$

where

I_1 = radiation intensity at point 1.

I_2 = radiation intensity at point 2.

d_1 = distance from the source to point 1 in cm.

d_2 = distance from the source to point 2 in cm.)

In this instance, d_1 is one meter and I_1 is value calculated from Table 2-1-1. I_2 is midscale value of ratemeter.

Once these calculations have been completed, check results by setting up source and placing detector at calculated distance from source (after properly turning detector on and adjusting it). Use arrangement shown in Figure 2-1-2 to avoid backscatter from ground - "ground" here implies that a level surface is chosen, such as a driveway or tennis court.

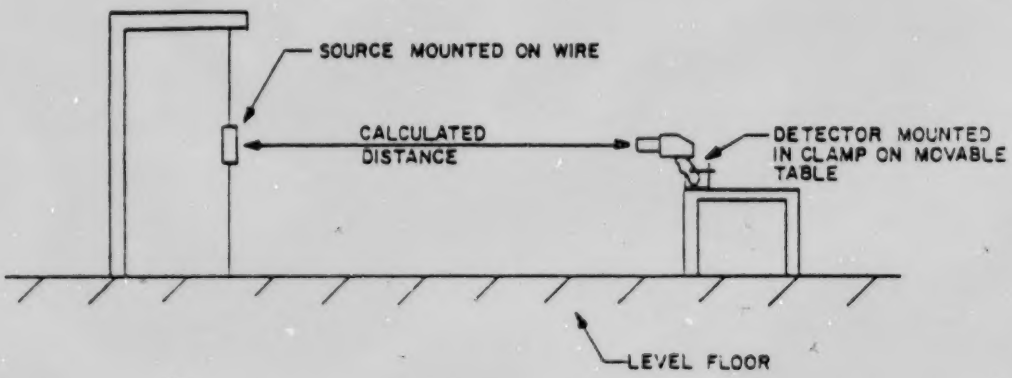

Figure 2-1-2. Preferred geometry.

Use all due precautions in handling source.

This procedure is used throughout the industry to calibrate ratemeter type detectors. Go over this procedure until it is firm in your mind.

DATA TABLE

DATA TABLE 2-1-1

1. Midscale values on lower two ranges of ratemeter:

   _____ mR/hr.

   _____ mR/hr.

2. Source material: _____.
   Its half-life: _____.
   Its activity, determined to be _____ curies on _____, 19____.

3. Source activity today: _____ curies.

4. From Table 2-1-1, midscale reading on lowest scale of ratemeter should occur at _____ meters from source; midscale reading on next range of ratemeter should occur at _____ meters.

5. If actual readings do not agree with calculations, check with instructor.

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Available by writing the Superintendent of Documents, U.S. Government Printing Office, Washington, D.C. 20402.

TEST

Read each question carefully. Be sure your responses give the specific information each question requires. Your answers should be brief.

1. What is the unit of exposure to radiation?

2. What do the values used to specify radiation exposure indicate?

3. Define exposure.

4. Define Roentgen.

5. How is the Roentgen related to the time of exposure?

6. What is the unit of energy absorption (absorbed dose)?

7. Define rad.

10. Complete the table below by converting the given quantities from rads to rems, using quality factors and the equation:

Rad x Q/F = Rem

Type of Radiation	Rad	Rem
Gamma	10 rad	_____
Beta	100 rad	_____
Fast Neutrons	10 rad	_____
Slow Neutrons	100 rad	_____

11. Very briefly, explain why the quality factor is different for different types of radiation.

_____

_____

12. For each radioisotope given on the table below, fill in the following: the Γ numbers, the dose rate at 1 meter from a 1 Curie point source; and the dose rate at 3 meters from a 1 Curie point source. (USE TABLE OF RADIONUCLIDES ON THE NEXT PAGE IN CALCULATING YOUR ANSWERS.)

Radioisotope	Γ Number	Dose Rate At 1 Meter	Dose Rate At 3 Meter
a. Iodine - 131	_____	_____	_____
b. Yttrium - 88	_____	_____	_____
c. Nickel - 65	_____	_____	_____

TABLE 2-1-1

GAMMA RADIATION LEVELS FOR ONE CURIE OF SOME RADIONUCLIDES*

NUCLIDE	Γ†	NUCLIDES	Γ†	NUCLIDES	Γ†
Actinium - 227	~2.2	Gold - 198	2.3	Potassium - 43	5.6
Antimony - 122	2.4	Gold - 199	~0.9	Radium -226	8.25
Antimony - 124	9.8	Hafnium - 175	~2.1	Radium - 228	~5.1
Antimony - 125	~2.7	Hafnium - 181	~3.1	Rhenium - 186	~0.2
Arsenic - 72	10.1	Indium - 114M	0.2	Rubidium - 86	0.5
Arsenic - 74	4.4	Iodine - 124	7.2	Ruthenium - 106	1.7
Arsenic - 76	2.4	Iodine - 125	~0.7	Scandium - 46	10.9
Barium - 131	~3.0	Iodine - 126	2.5	Scandium - 47	0.56
Barium - 133	~2.4	Iodine - 130	12.2	Selenium - 75	2.0
Barium - 140	12.4	Iodine - 131	2.2	Silver - 110m	14.3
Beryllium - 7	~0.3	Iodine - 132	11.8	Silver - 111	~0.2
Bromine - 82	14.6	Iridium - 192	4.8	Sodium - 22	12.0
Cadmium - 115m	~0.2	Iridium - 194	1.5	Sodium - 24	18.2
Calcium-47	5.7	Iron - 59	6.4	Strontium - 85	3.0
Carbon - 11‡	5.9	Krypton-85	~0.04	Tantalum-182	6.8
Cerium - 141	0.35	Lanthanum-140	11.3	Tellurium-121‡	3.3
Cerium - 144	~0.4	Lutecium-177	0.09	Tellurium-132	2.2
Cesium-134	8.7	Magnesium -28	15.7	Thulium-170	0.025
Cesium-137	3.3	Manganese-52	18.6	Tin-113	~1.7
Chlorine -38‡	8.8	Manganese-54	4.7	Tungsten-185	0.5
Chromium-51	0.16	Manganese-56	8.3	Tungsten-187	3.0
Cobalt-56	17.6	Mercury-197	~0.4	Uranium-234	~0.1
Cobalt-57	0.9	Mercury 203	1.3	Vanadium-48	15.6
Cobalt-58	5.5	Molybdenum-99	~1.8	Xenon-133	0.1
Cobalt-60	13.2	Neodymium-147	0.8	Ytterbium-175	0.4
Copper-64	1.2	Nickel-65	~3.1	Yttrium-88	14.1
Europium-152	5.8	Niobium-95	4.2	Yttrium-91	0.01
Europium-154	~6.2	Osmium-191	~0.6	Zinc-65	2.7
Europium-155	~0.3	Palladium-109	0.03	Zirconium-95	4.1
Gallium-67	~1.1	Platinum-197	~0.5		
Gallium-72	11.6	Potassium-42	1.4		

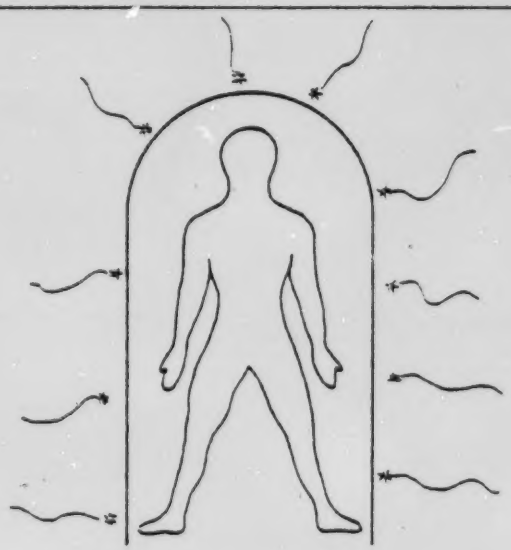
* Jaeger, R. G., et al., Engineering Compendium on Radiation Shielding, Vol. 1, (New York: Springer-Verlag, 1968), pp. 21-30.

† Γ = R-cm^2/hr-mCi or $\Gamma/10$ = R/hr at 1 m/Ci

‡ A Manual of Radioactivity Procedures [National Bureau of Standards Handbook No. 80 [Washington, D.C.: Supt. of Docs., U.S. Government Printing Office, Nov. 1961)], Appendix A, pp 137-140.

NUCLEAR TECHNOLOGY

MODULE 2-2 BIOLOGICAL EFFECTS OF RADIATION



RADIATION PROTECTION I

COURSE 2

COURSE MODULES

COURSE 2 RADIATION PROTECTION I

MODULE 2-1 Radiation Exposure and Absorbed Dose Values

MODULE 2-2 Biological Effects of Radiation

MODULE 2-3 Radiation Protection Standards

MODULE 2-4 Internal Dose Calculations

MODULE 2-5 Personnel Monitoring Devices, Ionization Types

MODULE 2-6 Personnel Monitoring Devices, Chemical Types

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INTRODUCTION

As noted in Module 2-1, "Radiation Exposure and Absorbed Dose Values," the effect of radiation can be caused by a direct or an indirect action. In this module, radiation effects, as they pertain to both the macroscopic and the microscopic view from the cellular level to the systemic level, are examined.

Radiation exposure is generally considered to be either acute or chronic. A single accidental exposure to a high dose during a very short period of time is commonly called acute exposure - effects of such exposure may appear immediately or at a later time. In continuous, long-term, low-level exposure, termed chronic exposure, results may not be apparent for many years.

Change in a biological system can be brought about by radiation through either (1) ionization of atoms composing the body or (2) excitation of these atoms to higher energy states.

The deposition of energy, by ionizing radiation, in any living organism results in biological damage. This damage can be beneficial (as is the case in destruction or alteration of cancer cells through radiation therapy) or detrimental (as is evidenced by radiation burns of normal tissue during radiation therapy, or in the alteration of chromosomes).

Biological injury through radiation can be classified as "somatic" or "genetic." Somatic effect is the effect on an individual. Genetic effect is the effect on future generations as a result of exposure to radiation of genetic material in this generation.

In studying biological effects of radiation, the following basic factors must be taken into consideration:

Biological variability -

Age.

Variation in radiosensitivity of different tissues.

Species variability.

Individual variability.

Total absorbed dose, chronic and acute -

Protracted exposure or dose administration over a long period of time (chronic).

Fractionation of dose - where exposure is spread out over a period of time in a series of doses, with periods of rest interposed between irradiations. Note: These rest periods influence recovery and repair.

Fraction of tissue irradiated.

Quality factor (QF) - a function of specific ionization and linear energy transfer, or the number of ions formed per unit distance along the path of an ionizing particle. Linear energy transfer (LET) is the energy locally absorbed per micron of path of ionizing radiation. For typical medical x-rays, a LET is ≈ 3 keV/micron.

Maximum permissible irradiation exposure varies according to the nature of the population group. A different value is set for the radiation worker as opposed to a lesser value assigned for the general public.

MODULE PREREQUISITES

The student should have successfully completed Course 1, "Radiation Physics," and Module 2-1, "Radiation Exposure and Absorbed Dose Values," and understand the principles covered. An understanding of general chemistry principles and calculations is recommended.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. List and discuss various ways that radiation can damage living body cells; describe the mechanisms involved.
2. Identify and explain theories of radiation effects.
3. Distinguish between direct and indirect radiation effects.
4. Explain why LET values differ with type of radiation and energy involved.
5. Discuss and outline a typical radiation syndrome.
6. Predict probable radiation effects for different rem ranges and types of radiation; indicate sensitive organs or tissues.
7. Explain difference in effects from internal versus external radiation exposure, and relationships of radionuclides to extent and type of injury produced.
8. Discuss difference between somatic and genetic effects of radiation; explain importance of each type.

9. Evaluate hazards of biological injury to general population versus injury to a radiation worker; explain benefit-versus-risk concept in the use of radiation.

UNDERSTANDING RADIATION EFFECTS

When biological consequences of absorbed radiation in man are discussed, it should be noted that radiation injury is not unlike any other variety of illness known to the medical profession. Once a cell has been damaged, the response is the same, regardless of the causative agent. Medical practicioners have long been faced with the awesome responsibility of diagnosing illnesses or conditions in man.

Generally, it is believed that changes in man that are deleterious have an effect that is characterized by an observable sign or symptom (referred to as a clinical symptom). These effects, whether caused by radiation or other agents, are, in all respects, similar. For example, if a lump is found on a patient's thyroid by a doctor, it would be very difficult, after being diagnosed as cancer, to prove whether the causative agent was radiation or some other deleterious substance. The medical practitioner would have to try to determine what, in fact, was the causative agent by eliciting a personal history from the patient. The doctor may learn that the individual worked in a chemical factory and had been inhaling some substance over a long period of time. The doctor may, then, because of expert advice or other means, make a determination that the cancer was caused by that substance. If, on the other hand, while taking the personal history, the doctor finds out that the patient had been working around radioactive substances, and specifically iodine, the doctor may then have reason to believe that the causative agent of this condition in this patient's thyroid may very well be due to radiation associated with the iodine

(because of the body's tendency to concentrate iodine in the thyroid).

In an attempt to study living organisms, from exposure to radiation to observation of effect, it must be assumed that a lengthy sequence of events may be involved. As discussed in the previous module, an assumption must also be made that changes take place at the molecular level. If a clinical sign or symptom is manifested in the patient; then it is known that an effect has, in fact, taken place and that a large number of ionizing events have occurred. It is also assumed that the theory of ionization relates to the complex living cells and that, with the deposition of any energy in the tissue, there is some effect, even if not observable. This radiation or ionization effect alters the structure or electrical charge in a cell. If the bound atoms are altered when the valence bonds are changed, as the radiation interacts with the electrons, then, on recombination, a different kind of substance may be found in that tissue (Figure 2-2-1).

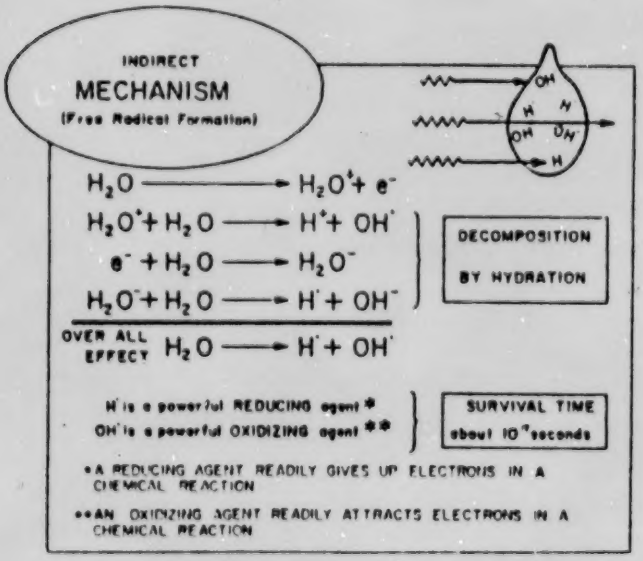

INDIRECT
MECHANISM
(Free Radical Formation)
OH
H
H
OH
OH
H
$H_2O \longrightarrow H_2O^+ + e^-$
$H_2O^+ + H_2O \longrightarrow H^+ + OH^-$
$e^- + H_2O \longrightarrow H_2O^-$
$H_2O^- + H_2O \longrightarrow H^- + OH^-$
DECOMPOSITION BY HYDRATION
OVER ALL EFFECT $H_2O \longrightarrow H^- + OH^-$
H is a powerful REDUCING agent *
OH is a powerful OXIDIZING agent **
SURVIVAL TIME about 10^{-5} seconds
* A REDUCING AGENT READILY GIVES UP ELECTRONS IN A CHEMICAL REACTION
** AN OXIDIZING AGENT READILY ATTRACTS ELECTRONS IN A CHEMICAL REACTION

Figure 2-2-1. Possible reactions resulting from deposition of ionizing radiation.

Also, if this substance is a direct product or effect of the breakdown product, they may react as follows:

Charged particles probably react immediately.

Noncharged particles react after a slight delay.

Chemical reactants (ions) form new molecules that may be breakdown products and, in many instances, are deleterious to the tissue in which they reside.

To discuss breakdown products, energy absorption, and molecular breakdown, an overview of the cellular level being considered is necessary. Also, some knowledge of cell structure is required. Cell structures contain a nucleus - a genetic bank - (Figure 2-2-2). In this diagram, a typi-

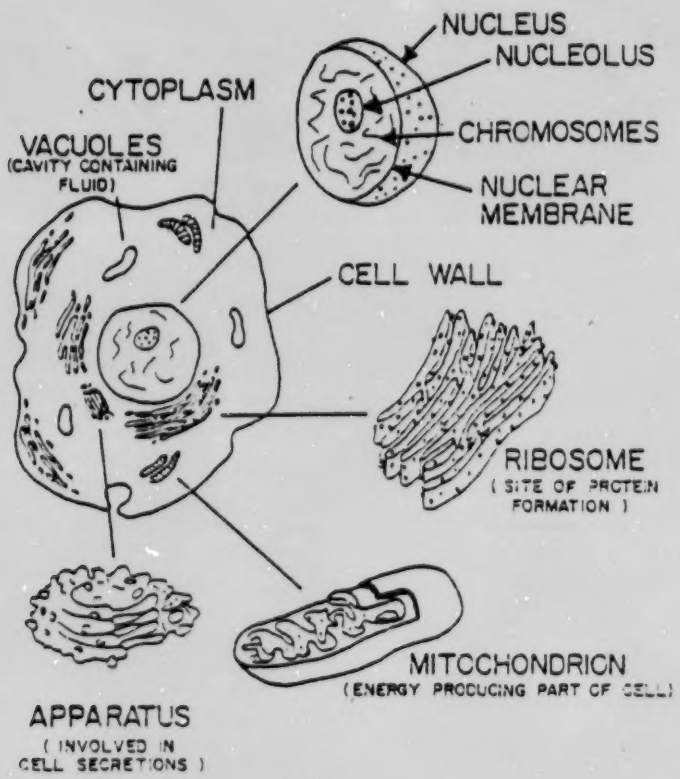


Figure 2-2-2. Diagram of a typical cell.

cal cell is shown to contain (in addition to the nucleus) cytoplasm, which is the cell body and most of the mass of the cell. The various components of the cytoplasm are involved in the cell metabolism.

The cell wall is the cell barrier between it and other cells. Because this cell is a living entity, it has a limited lifespan. During this lifespan, the cell must divide to produce daughter cells and carry out all the functions of a cell.

During the process of cell division (Figure 2-2-3), abnormal cells may not divide and, when they die, may not be replaced. If a cell is altered and daughters are genetically different, they may die before they reproduce.

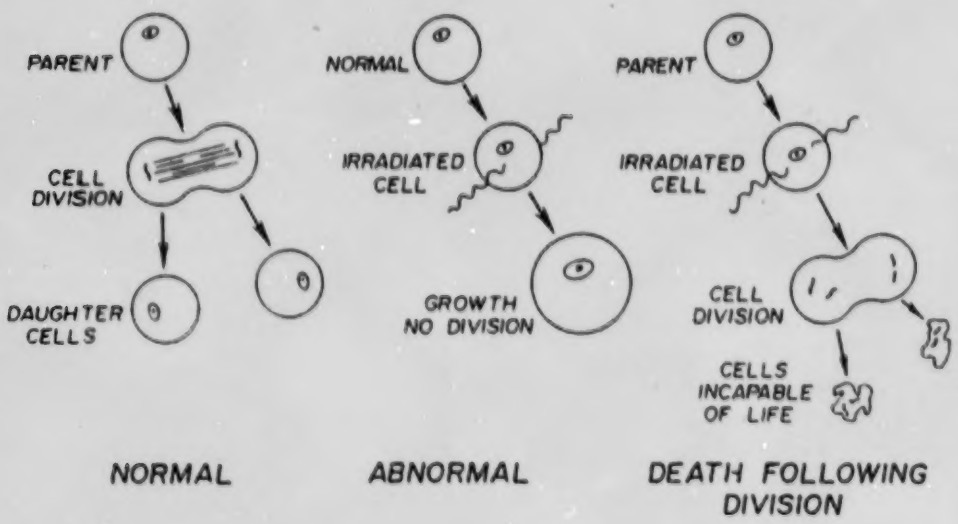


Figure 2-2-3. Cell division of normal and abnormal cells.

Death following division of cell may be due to absorption of energy in the form of radiation. In addition, com-

plex protein molecules, important in cell-wall formation, may allow cell-wall breakdown in irradiated cells because of the increased effect of radiation on the cell itself. Products necessary to the life of the cell may pass from the cell to other cells rather than being retained for the cell's own metabolism. When this happens, the result may be temporary or permanent injury to the cell (Figure 2-2-4).

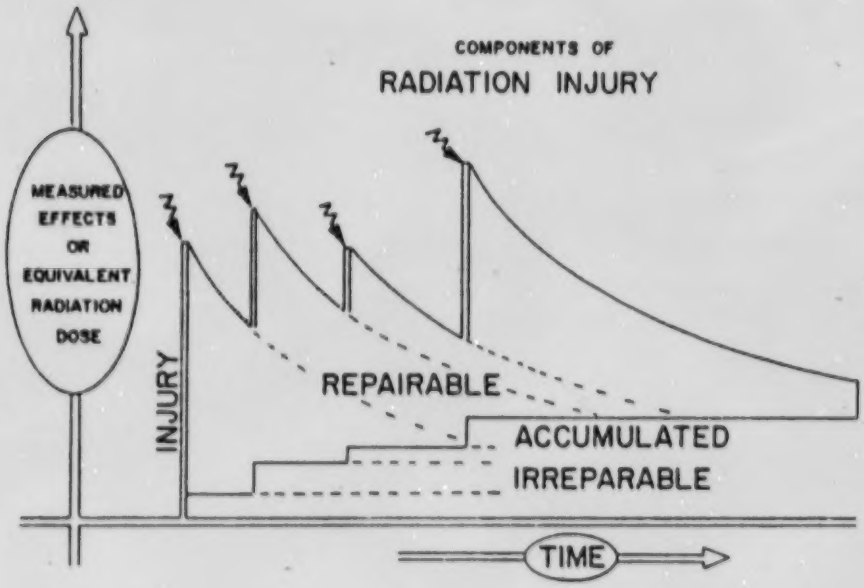


Figure 2-2-4. Recovery and exposure events.

Each time radiation causes some damage to a cell, a repair process is initiated; however, not all damage is repairable and, at some point in time after exposure, there will be an accumulation of unrepaired damage, which may be observed as a clinical symptom in the patient.

Biological effects result from the manner in which a cell interacts with energy. This interaction with energy

may be direct or indirect. It may be localized in the cell, or remote, through breakdown products. The time it takes for an effect to occur may be short or long, depending on the ions formed. The type of biological effect may be structural or functional. For example, structural effect is manifested by breakdown in a cell wall, which allows nutrients (necessary for cell metabolism and growth) to go from cell to cell, rather than being retained. Functional breakdown, inability of a cell to perform functions it is designed to undertake, also occurs. Biological effects may be reversible or irreversible. If, during the regular repair cycle, the cell repairs itself, or goes through the normal cell division, and becomes two new daughter cells, then the effect of the incident energy was reversible. If, in fact, the effects are irreversible, the cell will die, either before or following cell division. Acute effects are generally manifested and classified into three main classes. In order of increasing severity, these effects include:

Hemopoietic syndrome - that is, symptoms associated with blood-building tissues in the body.

Gastro-intestinal syndrome - that is, symptoms associated with digestive system.

Central nervous system syndrome.

Certain effects, common to all classes, include:

Nausea and vomiting.

Malaise and fatigue.

Increased temperature.

Blood changes.

LINEAR AND THRESHOLD EFFECTS

Linear and threshold effects of radiation are two contrasting explanations of the quantitative relationship between extent of damage and absorbed dose of radiation. If effect versus dose for various biological tissues is plotted, two relationship forms appear (Figure 2-2-5).

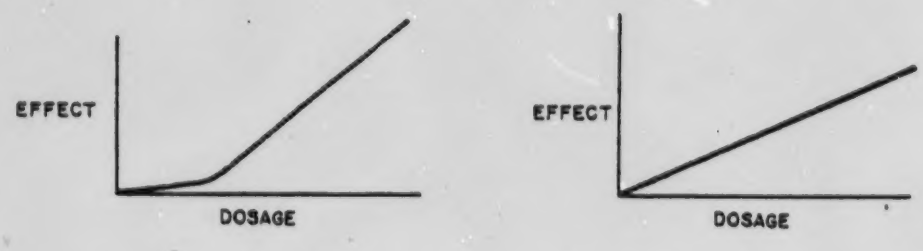


Figure 2-2-5. Types of radiation effects on biological systems.

A common biologic response (to most physical and chemical agents) becomes measurable after a certain minimum (threshold) dose is exceeded. The majority of acute effects of radiation are thought to be of this type.

As more refined means of measuring biological damage are developed, however, the linear relationship seen in Figure 2-2-5b may become more indicative of the true response. This "nonthreshold" curve, starting at the origin, indicates that a minimum dosage is not required to produce an effect. It is well established that genetic effects of radiation follow this type of dose effect curve. Considerable evidence

is accumulating that lifespan shortening and possible carcinogenic effects also follow the linear effect curve.

RADIOSENSITIVITY

Other effects may be seen in the radiation syndrome. The capacity of a body to recuperate from absorption of radiation depends upon the radiosensitivity of involved tissue. Response of a particular tissue to radiation determines whether or not the tissue is susceptible to that radiation. If a cell is very sensitive, probability of absorbing incident radiation and subsequently dividing normally during cell division is quite low. If, on the other hand, the tissue is rather insensitive, the likelihood of it absorbing radiation and having normal cell division is enhanced. Thus, cells not only differ in nature and appearance, but in response to radiation.

The law of Bergonie and Trebondeau states, ". . . radiosensitivity of a tissue is directly proportional to its reproductive capacity and inversely proportional to its degree of differentiation," which means that those cells which multiply rapidly, for example, cancers, are more sensitive than other tissue.

Highly specialized tissue, for instance, a nerve cell, is less susceptible than less highly specialized tissue. These effects, as stated in the law, become more evident with progression from cellular to a systemic biological effect study.

Older people are more susceptible than younger adult subjects. Also, very young people, in which cell growth and development is rapid, are more susceptible.

Some cells, such as lymphocytes, are very radiosensitive. On the other end (or other extreme), nerve cells, which are specialized and long lived, are less radiosensitive.

DIRECT AND INDIRECT EFFECTS

Generally, biological effects are produced by two mechanisms: 1) directly, by disassociating molecules following the creation of ion pairs (excitation) and, 2) indirectly, by production of breakdown products, probably hydrogen peroxide.

The gap in our knowledge is in the sequence of events between the primary effect and the gross biological effect observed long after irradiation. Acute radiation effects, resulting from a single accidental high dose in a relatively short period of time, are noticeable in a short time. Chronic radiation effects, a long-term, low-level overexposure, may not be noticeable for many years.

ACUTE EFFECTS

Acute effects, as they pertain to an individual's blood-forming, digestive and central nervous systems, are covered in the following discussion. The hemopoietic syndrome appears after an exposure dose of about 200 R*. The onset of symptoms, rather sudden, is characterized by nausea and vomiting within several hours after exposure. Malaise and fatigue

---

* Actual changes in blood may be detected at exposures as low as 25 R.

often accompany this condition, but there is no correlation between this effect and how much radiation has been absorbed. Epilation (loss of hair) is almost always seen, generally within two or three weeks. The blood-producing ability of bone marrow is impaired and, usually, a decrease in the white blood cell count appears after about one day. The loss of white blood cells increases the likelihood of secondary infections. Complete destruction of bone marrow generally occurs in the range of 400 to 600 R, however, the marrow can be regenerated if the patient survives the other effects of the exposure. Irreversible destruction of the marrow almost always accompanies an exposure that is equal to or greater than 700 R. If a very low white blood count is seen within the first few days after exposure, it suggests that death will probably ensue.

If an exposure greater than 1,000 R occurs, a gastrointestinal syndrome appears. It is a direct result of damage to the lining of the intestinal walls. At these levels of exposure the damage is so great that the lining is literally sloughed off. All of the preceding blood-syndrome changes are seen, and, in addition, severe nausea, vomiting and diarrhea begin soon after the radiation has been absorbed. At this dose, onset is extremely abrupt, and death is likely to take place within one to two weeks after exposure. If the exposure is greater than 2,000 R almost all of the other organs of the body become involved. Above about 5,000 R, the central nervous system also is severely damaged. The patient will very quickly become unconscious (within minutes) and die within a few hours or days.

In considering these syndromes, it is helpful to look at an actual case. A white male was exposed to mixed neutrons and gamma radiation for a period of approximately 200

microseconds. His estimated dose was 4,400 rads. Very quickly, he became disoriented (in approximately one-half minute) and in ten minutes he was unconscious and in shock. Rigorous medical therapy was started, but the patient passed copious quantities of fluid into the gastro-intestinal tract. Effects of the dose were manifested in unconsciousness and shock almost immediately. Because the gastro-intestinal tract was involved, the patient literally dehydrated because of the inability of the gastro-intestinal tract to reabsorb fluids back into the system. All symptomatic treatment failed, but, because of blood transfusions and other electrolytes being pumped into his venal system, he was kept alive for approximately a day and a half.

DELAYED EFFECTS

Delayed effects of radiation may be due to a single, large overexposure, or to continuing low-level exposure. Also, they may be due to an external or an internal deposition of energy. Internal exposure is generally caused by inhalation of radioactive particles, or by ingestion along with food or drink, or by absorption through wounds in the skin. If radioactive material is deposited internally, it likely will combine with normal nutrients and be systematically absorbed within certain organs or tissues.

If the dose referred to in the delayed effects is internal, the adjectives acute and chronic, as ordinarily used, are not directly applicable. When the delayed effect is investigated, the word "cancer" comes to mind. It is common to refer to cancers as an occupational syndrome of radiation workers. It is true that overexposure to radia-

tion may cause certain organs, because of their sensitivity, to manifest the incident radiation in the long-term effect as cancer. Any organ may be involved.

Certain organs and tissues seem to be more likely to develop cancer than others. Cancer is seen most frequently in skin, blood system, bone and thyroid gland. Tumor induction in man is rather long, on the order of five to twenty years after exposure, which is also typical of cancers induced by other agents.

As already stated, a particular cancer cannot be identified as having been caused by a specific exposure to a chemical or radiation, although circumstances may point in that direction. For example, skin carcinoma was first associated with exposure to x-rays. In 1900, a physician documented his x-ray exposure and his symptoms as going from erythema (reddening) and itching to hyperpigmentation, ulceration, neoplasia and, finally, death from metastatic carcinoma. This individual's case, from erythema to death, took a little over nine years.

Another delayed effect of radiation is leukemia. Leukemia is generally associated with whole-body exposure. Records indicate that the incidence has been greatest with radiologists. As more and more people are occupationally exposed, there may be other groups to examine. An increased incidence of leukemia seems likely in children who have been irradiated in utero. Pregnant women who receive x-rays, diagnostic or therapeutic, run a greater risk of bearing children who will develop leukemia than a similar group bearing children from nonirradiated fetuses. There is now reason to believe that the incidence of leukemia is much greater in those Japanese who were within 1,500 meters of ground zero in both Nagasaki and Hiroshima than the rest of the

population. This particular group of people has been studied since both bombs were detonated in 1945, along with their progeny; there is every reason to believe that they and their heirs will be studied in the future. A threshold effect for leukemia has not definitely been proved, but a general consensus of opinion among radiobiologists is that if a threshold exists it probably lies between 0 and 460 rem.

Another delayed effect is bone cancer, probably caused by internally deposited radionuclides. The first deaths from this were recorded in the 1920's (in the radium paint industry). These patients, because of their occupation, ingested large amounts of radium. (It was their common practice to roll their paint brushes on the tips of their tongues to get very fine points, which enabled them to paint radium onto watch dials. As patients, they often manifested signs of anemia, accompanied by bone cancer, usually of the jaw. (Radium is a tenacious bone seeker, and significant amounts were discovered in workers even after a period of 35 years had elapsed.) It was many years later, after the manifestation of the cancer of the jaw, that the industry policed itself and set up safety standards. Radium is one of several radionuclides that are bone seekers, and bone seekers are so toxic because they damage the blood-forming tissue in the bone marrow.

Lung cancer is another delayed effect of incident radiation. Lung cancer, which is caused primarily by inhalation of radionuclides, is common among certain miners. Within fifteen years of being subjected to about 30 picocuries per milliliter of air containing radon and its daughters, radium A and radium C, mine workers were found to have a significantly higher indicence of lung cancer than other

workers. When mice, under controlled conditions, were exposed to one nanocurie per milliliter of radon, these findings were confirmed.

Another effect attributable to radiation is life shortening. In certain animal studies, those groups exposed to radiation died earlier than those not exposed. No such clear-cut effect has been seen in man. Comparison of the life duration of medical radiologists with that of other medical personnel suggests conflicting implications. Some data lead to the conclusion that radiation shortens life, while in other instances no statistically significant difference in life expectancy was noted. Generally speaking, however, the scientific community believes that low-level overexposure to radiation will cause some degree of life shortening.

Another delayed effect is cataracts. Cataracts may be due to chronic or acute exposures. These exposures cause opacity of the lens of the eye (opacities are cataracts). Generally, the catarogenic dose to the lens of the eye is on the order of 500 rads for beta or gamma. Cataracts have been reported for doses of 200 rads for mixed gamma-neutron radiation. It is believed that the threshold dose may lie between 15 and 45 rads for neutrons. There have been no reported cases caused by x-ray to the occupationally exposed.

In general, biological effects of irradiation relate to the type of radiation encountered (Table 2-2-1).

INTERNAL AND EXTERNAL SOURCES

It is necessary to take into account that biological

TABLE 2-2-1
DOMINANT BIOLOGICAL EFFECTS FOR
DIFFERENT TYPES OF RADIATION

RADIATION TYPE	QF	BIOLOGICAL EFFECT
X, gamma, beta rays (photons & electrons)	1	Whole body irradiation, blood system is critical
Photons & electrons below 10 keV, low energy neutrons and protons	2-3	Whole body irradiation, blood system effects are secondary - outer surface is critical
Fast neutrons and protons 0.5 MeV -	10	Whole body irradiation, cataracts are critical
Natural alpha particles (i.e., radium emitted)	10	Cancer induction
Heavy recoil nuclei, fission particles	20	Cataract formation

effects vary according to whether radiation is delivered from an external source or from a source within the body. X-rays and gamma rays may cause biological effects throughout the entire body when delivered from external sources. On the other hand, radionuclides emitting beta or alpha particles may cause much greater biological damage to a particular organ or organs when their energy is expended from an internal source.

If a radioactive element is taken into the body by ingestion, inhalation, or absorption through a break in the skin, it may be deposited in various organs of the body. Chemical characteristics of the radioactive material will determine where and to what extent the material is deposited and concentrated, and its excretion rate. For example, radioactive strontium and radium have chemical properties anal-

ogous to calcium; thus, they are deposited in bone in a manner similar to that in which calcium is deposited.

Solubility and particle size of a radionuclide determine how much of it gains access to circulating blood, and to critical organs. Once a nuclide is deposited in tissue, it is extremely difficult to increase its natural elimination rate.

The amount of damage imposed by an internally located radionuclide is governed by the concentration of the radionuclide in the tissue or critical organ, by half-life of the radionuclide, by elimination rate of that chemical species and by the type of radiation emitted by the nuclide.

To determine the magnitude of QF values, the student is referred to extensive tables developed by the International Commission on Radiation Protection. These tables list maximum permissible concentrations (MPC) in air and water and maximum permissible body burdens (MPBB) for all radionuclides in common use.

Basic rules that apply to accumulated dose for radiation workers are set out in NCRP Report 39. Legal limits are set forth in the Code of Federal Regulations, particularly 10CFR20, and various state codes.

These limits are based on recommendations of the NCRP. When adopted, these recommendations become laws or regulations. The following legal limits are for radiation workers:

Rem per calendar quarter:

Organ	Rem/Qtr
Whole body; head and trunk; lens of eye; gonads	1 1/4
Hands and forearms; feet and ankles	18 3/4
Skin of whole body	7 1/2

Under certain criteria these limits may be exceeded:

a. During any calendar quarter the dose to whole body shall not exceed 3 rem.

b. Dose to the whole body, when added to occupational dose already received, shall not exceed 5 (N - 18) rem, where N is the individual's age at last birthday.

c. Individual's complete occupational dose history is known.

For internal exposure, the maximum permissible concentration (MPC) of radionuclides in air and water is calculated on the basis of an average of 15 rems for most individual organs of the body, 30 rems for the thyroid or skin and 5 rems for the gonads or whole body. For bone seekers - radionuclides that concentrate in bone - MPC is based on distribution of radionuclide in the body, on QF and on a comparison of energy release in the bone with energy release delivered by a maximum permissible body burden of 0.1 mg ^{226}Ra plus daughters.

An accidental or emergency dose of 25 rems to the whole body, or a major portion thereof, occurring only once in a lifetime of the person, need not be included in determination of radiation exposure status for that person.

SOMATIC AND GENETIC EFFECTS

The somatic effects of radiation are those effects on an individual's body cells that affect the individual's biosystem. Genetic injuries manifest themselves in the offspring of irradiated individuals and may not be apparent

for many generations. Their detrimental effects propagate throughout the population as exposed individuals mate and produce offspring.

Somatic effects of radiation on the vital functions of body cells may be listed as follows:

Mobility may be restricted, suppressed or remain unchanged.

Reproductive power may be restricted or suppressed. Sometimes, irradiation stops cell division. In other instances, irradiation results in abnormal cell division.

Growth may be restricted or suppressed. Effect on growth usually becomes manifest after a latent period.

In metabolism, fatty, mucoid (protein) or amyloid (starch) degeneration may follow exposure to irradiation. Metabolism of an irradiated cell may be altered and approach that typical of an abnormal cell, such as a cancer cell.

Specific cell functions, such as gland secretion, may be suppressed or restricted.

Long-term somatic effects include 1) cancer and other malignancies, 2) lifespan shortening, 3) impaired fertility and 4) cataract formation. Long-term effects are usually associated with a low-level chronic exposure to radiation.

Acute radiation exposure also can result in long-term effects after a latent period that follows an acute radiation syndrome.

While most of the evidence of long-term effects has been derived from studies of animal populations, some corroborating evidence is available from large-scale population studies of humans subjected to medical x-ray exposures

and from studies of the Hiroshima and Nagasaki bomb victims and their children.

In terms of effects of radiation on the embryo, the most radiosensitive period in the human prenatal life occurs from conception through about day 40. In normal embryological development during the first 20 days, even low levels of radiation produce an increase in spontaneous abortions.

Radiation exposure after the embryo has become fixed in the uterus (20-40 days) may produce any of the following:

Prenatal death.

Neonatal death.

Abnormalities (stunted growth, mental retardation, etc.).

After about six weeks, higher levels of exposure are necessary to produce abnormalities.

The genetic effect of radiation is due to deposition of radiation energy within the genes (DNA) of the sex cells (eggs or sperm) of an organism. This alteration of hereditary information conveyed by the genes is known as mutation.

All genes are subject to mutation. This process occurs at definite but normally low rates in any population. The spontaneous mutation rate in man is on the order of 1/100,000 per gene per generation or one in every ten eggs or sperm. Some of the causal agents of these spontaneous mutations include:

Natural high-energy radiation (cosmic rays).

Chemical agents.

Errors in gene duplication.

The following working assumptions have been derived from studies of genetic effects of radiation on man and other animals:

Radiation-induced mutations at any given dose rate increase in direct linear proportion to genetically significant dose.

Mutations, once completed, are irreparable.

Almost all observed effects are harmful.

Radiation-induced mutations are, in general, similar to naturally-occurring mutations.

There is no known threshold dose below which some effect may not occur.

MAXIMUM PERMISSIBLE RADIATION EXPOSURES

Maximum permissible irradiation is determined for two conditions:

That which, in the light of present knowledge, is not expected to cause appreciable injury to a person at any time during life.

That which is not expected to produce excessive genetic damage.

If no threshold exists and if there is no repair of damage (the safest assumption), the first condition depends upon the accumulated radiation received during an individual's lifetime. The second condition depends upon accumulated radiation received up to the time the individual produces offspring.

In setting permissible levels, it is important to keep genetic damage in the whole population low, but it is also important to allow for beneficial uses of radiation. Thus, an attempt is made to balance the values of radiation against its dangers by setting up different values for separate situations and separate populations.

Thus, the general population with its large and essential genetic pool and with no special controls on radiation it receives is allotted a lower maximum (0.1 of the permissible occupational level). Thus, for the general population, the radiation dose should not exceed 0.5 rem/yr and, for the average yearly whole body, exposure should not exceed 0.17 rem/yr (1/3 the individual dose), while the thirty-year total should not exceed 5 rem for the individual.

Due to the selected population, voluntary assumption of risks, controlled environs and the small genetic pool affected, the radiation exposure value for radiation workers is a whole body dose of 5 rem/yr, with some allowance for a once-in-a-lifetime emergency exposure beyond the guidelines.

The Federal Radiation Council agrees with the guidelines issued by the National Council on Radiation Protection for occupational exposure: this dose of ionizing radiation is not expected to cause an appreciable body injury to a person at any time during a lifetime.

It is also important to note that the occupationally exposed are more carefully monitored for daily and weekly assessment of cumulative exposure. In addition, physiological changes are carefully observed through periodical physical examinations.

Finally, the radiation worker is further protected through worker's compensation laws, which provide for monetary compensation of radiation-induced injury.

EXERCISES

None.

MATERIALS

None.

PROCEDURES

This module is presented as a series of lectures and discussion groups with assigned textbook and reference readings.

A library project on one of the epidemiological studies dealing with radiation-exposed populations should be the assigned activity. Student should write a paper, discussing the population studied, reasons for the study, and describing significant conclusions of the study.

DATA TABLE

None.

REFERENCE MATERIALS

Cember, H. Introduction to Health Physics. Elmsford, NY: Pergamon Press, 1969.

Hendee, W.R. Radioactive Isotopes in Biological Research. New York: John Wiley & Sons, 1973.

Lapp, R.E. and Andrews, H.L. Nuclear Radiation Physics. New York: Prentice Hall, 1972.

National Council on Radiation Protection and Measurements (NCRP). "Basic Radiation Protection Criteria," NCRP Report No. 39. Washington, D.C.: NCRP, 1974.

TEST

Read each question carefully. Be sure your response supplies the specific information each question requires. Make your answer brief. Use the backs of the test if you need additional space.

1. Briefly explain the threshold effect theory of radiation damage.

2. Briefly explain the linear effect theory of radiation damage.

3. What is the difference between a structural and a functional biological effect of radiation on cells?

4. What is the difference between somatic and genetic biological injury due to radiation?

5. What is the relationship between radiation's quality factor and the dose absorbed? What is the relationship between quality factor and linear energy transfer?

_____

_____

_____

_____

6. List the three classes of acute effects of radiation.

a. _____

b. _____

c. _____

7. List the four symptons common to each class of effects of radiation

a. _____

b. _____

c. _____

d. _____

8. How do reproductive capacity, lifespan, and degree of differentiation affect a given type of tissue's radiosensitivity?

_____

_____

_____

_____

9. What is the difference between an indirect biological effect and a direct biological effect of radiation?

_____

_____

_____

10. List three factor that determine the amount of damage an internally located radioisotope will have.

    a. _____

    b. _____

    c. _____

11. List two possible short-term somatic effects that radiation has on cells.

    a. _____

    b. _____

12. List one possible genetic effect of radiation.

    a. _____

13. Why is the maximum permissible exposure for radiation workers different from the maximum permissible exposure for the general population?

    _____

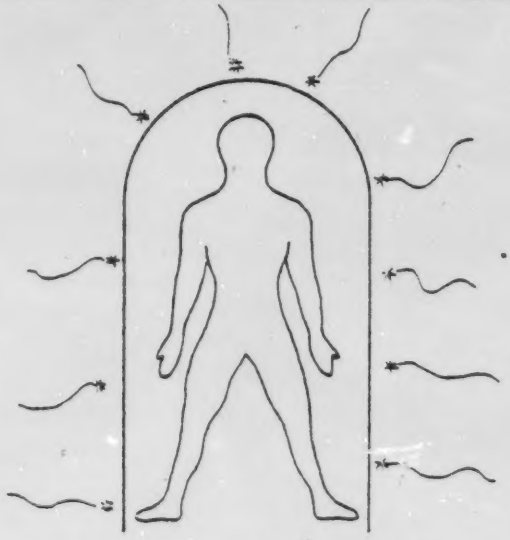
    _____

    _____

    _____

NUCLEAR TECHNOLOGY

MODULE 2-3 RADIATION PROTECTION STANDARDS



RADIATION PROTECTION I

COURSE 2

COURSE MODULES

COURSE 2 RADIATION PROTECTION I

MODULE 2-1 Radiation Exposure and Absorbed Dose Values

MODULE 2-2 Biological Effects of Radiation

MODULE 2-3 Radiation Protection Standards

MODULE 2-4 Internal Dose Calculations

MODULE 2-5 Personnel Monitoring Devices, Ionization Types

MODULE 2-6 Personnel Monitoring Devices, Chemical Types

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INTRODUCTION

Standards and regulations for radiation protection have undergone considerable change, and are apt to continue to evolve over a considerable period of time. Each new bit of research on the biological effects of radiation on the human body has the potential to cause some modification in the standards.

Past changes in the standards have resulted from the fact that there is insufficient hard information about the real effect of various types of radiation on the human body. This general lack of detailed information arises because our present information comes mainly from accidents involving human subjects or experiments conducted on animals, plants and so on. Radiation effects are a multifaceted problem that does not have simple answers.

For example, it is generally agreed that a one-shot, whole-body dose of 750-1000 rems, given in a short period of time, will kill anyone within a sixty-day period; if the dose is reduced to 450 rem, about half the people exposed will die within sixty days (LD50/60).

There are still questions about repetitive small doses of radiation. Are the effects cumulative? Or can the rebuilding mechanisms of the body initiate recovery? The accepted theory of cumulative damage and body recovery was still under examination a few years ago. After the debate ended, it was decided that the existing threshold concept was not sufficiently conservative. Current thinking is that all exposure of people to radiation should be "as low as practical."

Information used to arrive at this conclusion is not sufficiently clear or hard to establish a firm conclusion.

If more data disproves the conclusion and it is found, for example, that small amounts of radiation, like vitamins and minerals, are helpful to the body, another change may evolve in radiation standards.

So, standards and regulations are based on current best thinking and, as such, are evolutionary. They are believed by many to be highly conservative, although others disagree. Certainly, the student should be aware of radiation standards. Those standards used every day should be committed to memory.

In this module, radiation standards of the "to-be-memorized-as-soon-as-possible" type are covered carefully; the less urgent ones are discussed from the standpoint of, "what it is?" "where can it be found?" and "when is it needed?"

Because of a variation in state laws regarding radiation, and a variation in the application of federal laws, the emphasis in this module is on federal laws and regulations.

MODULE PREREQUISITES

Student should be familiar with subject matter of preceding modules in this course.

Note: The content of this module is basically legalistic. A copy of Title 10 of the Code of Federal Regulations is a must reference to have at hand while studying this module.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. State permissible dose equivalents for the following:
   - Whole-body dose.
   - Dose to extremities.
   - Dose to skin.
2. Calculate exposure allowances and working times, given permissible dose information.
3. List and discuss required steps whenever allowable radiation limits are exceeded.
4. Define and discuss the following:
   - Accumulated dose equivalent (ADE).
   - Title 10, Code of Federal Regulations, Part 20.
   - Handbook 69.
5. Calculate ADE's for persons of various ages.
6. List and discuss three subcategories of radioactive materials, including that part of 10CFR20 by which each is regulated.

HISTORY OF STANDARDS

In the beginning of radiation related activities, there were no standards or regulations. There are many true horror stories of how people were injured and killed because of the lack of intelligent guidelines to follow. Such fatalities and injuries led to the adoption, enforcement and continued evolution of standards.

There was a time when shoe stores had x-ray (fluoroscope) machines to aid in fitting shoes properly. A customer merely put the new shoes on, put the feet into the slot, pushed the button and observed the toes inside the shoes through a viewer. Kids could stand for hours wiggling their toes and watching them. Great fun! Unfortunately, such uncontrolled use of radiation has a potential for harmful effects. As a result, such machines have disappeared.

Early standards permitted exposures that today are considered unacceptably hazardous. For example, the Second International Congress of Radiology (Stockholm, 1928) established permissible doses of x-rays based on the exposure necessary to redden the skin!

The National Committee on Radiation Protection (NCRP), organized in 1946 and still in business today, continually updates information on radiation limits. In July of 1959, the NCRP published its Handbook 69, "Maximum Permissible Body Burdens and Maximum Permissible Concentration of Radionuclides in Air and in Water for Occupational Experience." In August, 1970, the National Committee on Radiation Protection published Report No. 39, "Basic Radiation Protection Criteria." (The latter document was used to prepare

this module.) Both documents must be mentioned frequently as the story of standards development is unfolded. Frequent reference must be made also to Title 10, Code of Federal Regulations, Part 20, "Radiation Protection Standards."

How is a standard established when the researcher is not sure what is harmful and what is not? Obviously, the goal is a standard as conservative as possible and one such that, even if it is wrong, no one will be harmed. Such standards have not always been achieved, but are sought to the best of the researcher's ability, based on available information. And, as additional data becomes available, modifications are apt to occur.

ACCUMULATED DOSE EQUIVALENT

The regulation that governs accumulated dose equivalent (ADE) is found in Title 10 of the CFR. (The first subheading of the code is the title; the next is the chapter.) Title 10 has only one chapter, divided into parts. Part 20 deals with radiation protection. (The student must become familiar with Part 20, normally referred to as 10CFR20.) These parts are further subdivided into sections, and the rules governing ADE are found in 10CFR20.101(b)(2).

This section of 10CFR20 says, in essence, that the maximum occupationally accumulated dose cannot exceed 5(N-18) rem. (N indicates a person's age in years as of last birthday.) This standard implies that the total occupational dosage received up to any age must be less than the ADE ("occupational" means that medical or dental x-ray doses

received as a patient do not count, but dosages received in previous jobs do count).

Regulation 10CFR20.101(b)(3) also requires that each employer keep a record of exposure and past history of exposure for each employee on an NRC 4 form. Such information should be available from a person's previous employers, except possibly for companies that went out of business before 1967 (how this latter situation is handled is discussed later).

For an employee 18 years of age or younger, the ADE is zero - the occupational dosage allowed is ZERO. But other limits may apply. Actually, 10CFR20.104 allows people under 18 years of age a radiation dose up to 10% of the limit for persons age 19 years. Thus, ten percent of the ADE for a 19-year-old person is 500 millirem/yr.

As stated in 10CFR20.6, if there is doubt about interpreting the regulations, the federal government reserves the right to interpret the rules. Therefore, the nearest Compliance Officer should be called when in doubt. Five Compliance Offices are situated around the country with current addresses and phone numbers listed in 19CFR20, Appendix D. In most cases, a Compliance Office will provide prompt answers to any questions.

At each birthday, an individual's allowed personal ADE increases by 5 rem (except for those under eighteen). Thus, if a new employee is 25 years old next month and has never worked in a radiation environment, that employee's allowed ADE is 30 rem. If, on the other hand, an individual has worked in the nuclear industry for two years and the NRC 4 form indicates that 10 rems have been received in those two years, that individual can accumulate (30 - 10) 20 rem before exceeding the allowed ADE.

APPLICATIONS OF ADE

Suppose work is required in a high-radiation area. It might seem practical to hire a worker, use up that worker's entire ADE in a short time, then replace the worker. But, clearly, such practice would be unethical and illegal. In addition to the unethical aspects of such a policy, it violates the "as low as reasonably achievable" philosophy, and it violates 10CFR20.101(b)(1). This rule states that, with a known history (NRC 4 form), the maximum allowable rate of dose is limited to 3 rem per calendar quarter.

What is a calendar quarter? The answer is given in 10CFR20.3(a)(4), which states, briefly, that there are four such terms per year, and each is normally 13 weeks long. Some freedom in interpretation is allowed, however, and there may be two 12-week terms and two 14-week terms in a calendar year, starting any time between January 1 and March 31. This flexibility in definition allows some shifting to avoid a pileup of record-keeping. For example, the results of the film badges sent out for processing are not available until January 20, so it would be reasonable to start the quarter on February 1 instead of January 1.

If there are no records (NRC 4 form) on a new employee who was employed in radiation work, say, for a year, fifteen years ago, then the safe assumption to be made is that for that year the employee received 3 rem/quarter, or a total of 12 rem. For those years in which the employee was not working with radiation, the occupational exposure is, of course, to be taken as zero rem/quarter.

Consider the same "new hire" employee who will be 25 years old next month and who has an allowed ADE of 30 rem. By assuming the employee has never received a prior occupa-

tional dose, if the employee is now exposed at the maximum rate, how long will it take to reach the ADE?

To answer this, in addition to birth date, it must be known when the quarters end. Assume that the quarter ends next week. Between now and the end of the quarter, according to the safe assumption made in the above paragraph, the employee can accept an exposure of 3 rem. Assume that is the exposure. That leaves an uncommitted ADE of 27 at the start of the next quarter. Then, the employee can accept 3 rem in the next 13-week period and, at the end of that quarter, the balance would be 24 rem. But, this calculation is not correct. The employee had a birthday, and another 5 rem ADE was allowed. So, at the end of the next quarter, by assuming that the employee again took a 3-rem exposure, the unused ADE would be 29 rem (27 + 5 - 3).

Actually, if the employee's exposure continues at the rate of 3 rem/quarter each year, the employee loses seven rem (12 - 5) from the unused balance each year. Starting with the 25th birthday, with an allowed ADE of 29 rem, it takes something like 29/7, or a little over 4 years, to reach an ADE of 29.

Once the allowed ADE is reached, no more exposure is allowed until the next birthday, when the unused ADE increases to 5 rem. Then, the employee could receive 3 rem the first quarter and 2 rem in the next, reducing the allowed ADE back to zero. Thereafter, the employee would not be allowed to work with radiation until the next birthday, and so on.

In actual practice, the 3 rem/quarter limit is rarely approached, and then only in extreme circumstances, because it is more reasonable to keep an employee working at a job all year long. Assuming two weeks vacation per year, or one-half week of vacation per quarter, the employee would

work twelve and a half weeks per quarter. If radiation is limited to 100 millirem/week, the maximum dose is 1250 mrem, or 1.25 rem/quarter. For a year, the four-quarter total is 5 rem. Thus, for a worker whose unused ADE is zero, working to a limit of 1.25 rem per quarter neither increases nor decreases the balance, and allows the employee to work continuously.

This 1.25 rem/quarter limit is the lowest one specified in 10CFR20. Anything lower or more restrictive than that, such as 100 mrem/week or 20 mrem/day are local limits, usually imposed to keep better control of exposure to ensure that people continue at their jobs all year. Special authorization is usually required to exceed these local limits, and planned exposures in excess of 3 rem/quarter (or the 1.25 rem/quarter limit if that is applicable) should not occur.

Reference is sometimes made to a so-called once-in-a-life dose. An antecedent to 10CFR20 (a document called Handbook 69) was published in June, 1959, by the National Bureau of Standards and contains the recommendations of the National Committee on Radiation Protection (NCRP). (Most of its recommendations were carried into the revision of 10CFR20 as of January 1, 1961.) That handbook recommended that once-in-a-lifetime doses be ignored as far as ADE accounting was concerned. That is, the ADE record should show that a person had taken a once-in-a-lifetime dose and was never to do it again.

Before radiation standards were defined, the occurrence of certain radiation incidents established that, at dose rates in excess of 1,000 rem/year, noticeable injury could occur, but, below about 290 rem/year, no gross effects were discerned. With this in mind it was believed that 25 rem

might be easily absorbed as a one-shot dose without difficulty. So, 100 rem became the recommended once-in-a-lifetime dose which might be taken in an acute emergency, such as to save a life.

In less urgent emergencies, the 25 rem whole-body dose still applies. But, everyone in the industry agrees that, to save a life, they would take 100 rem. From a legal standpoint, however, this would be a violation of both the 3-rem/quarter limit and the ADE limit, unless the person taking the dose is more than 45 years old. In any event, the incident would have to be reported in accordance with 10CFR20.405.

Section 10CFR20.401-20.408 details the records and reports required by NRC for individuals, incidents, plants and so on. In addition, there may be applicable state reporting requirements. And, in the case of nuclear reactors, additional requirements are usually included in the plant's license Appendix A, called the Technical Specifications, or "Tech Specs."

In addition to whole-body irradiation, the only case discussed so far, there are limits for so-called critical organs. These organs include those most sensitive to radiation damage under the specific conditions being considered. They often include the blood-forming organs, the skin, lenses of the eyes, and the gonads. (The blood-forming organ is the bone marrow, so the critical organs can include the head and trunk of the body.) For irradiation from internally deposited sources, alone or combined with external sources, additional limiting organs include the gastro-intestinal tract, lung, bone, thyroid, kidney and muscle.

A separate limit, in effect for the skin of the body, is set at 30 rem/year (no ADE, just an annual limit). This

limit applies when a radiation environment exists that is predominately of nonpenetrating radiation (alpha, weak beta, or grenz* radiation). If goggles or safety glasses are worn to protect the eye lenses, then skin dose limits apply.

For the extremities, separate limits have been established:

Hands and feet - 75 rem/year and 18 3/4 rem/quarter. (These limits are directed toward glove-box applications; i.e., where dosage is entirely to the hands.)

Forearms - 30 rem/year and 7 1/2 rem/quarter.

It is true that the bones of the extremities contain marrow, but if that marrow were to completely cease producing blood (and it should not at this dose limit), the body would not suffer significantly; marrow in head and trunk bones is sufficient to produce enough blood.

NUCLEAR MATERIALS STANDARDS

This subject area has many aspects, all directed toward the control of radioactive materials in their various forms and/or from their various sources. The main subcategories include:

* Grenz ray: An x-ray produced at long-wavelength end of x-ray spectrum, involving wavelengths on the order of 1 to 10 angstroms, by using special x-ray tubes that operate at voltages from 5,000 to 15,000 volts. These rays are also called "soft x-rays," as opposed to hard, high-energy x-rays.

Source materials.
Special nuclear materials.
Byproduct materials.

Source materials are not those materials commonly referred to as sources in radiation protection work. Rather, they are defined in 10CFR40 as, essentially, uranium and thorium ores, the raw materials for reactor fuels. Special nuclear materials are defined in 10CFR70 as, uranium enriched to greater than 20% ^{235}U, ^{233}U and plutonium.

All other radioactive materials are classed as byproduct materials. Extensive regulations governing byproduct materials are contained in 10CFR36. The reason for the above delineation is that the uses of these materials are many, ranging from luminous watch and clock dials through radiographic sources to medical applications.

Source materials may be bought, sold, used, and so on, only by parties or organizations licensed under Part 40, except for two possible exemptions. First, if the quantity of material involved is below a certain minimum amount (10CFR40.13) and, second, if a special exception is granted (10CFR40.14). The remainder of Part 40 deals with the process of obtaining the type of license required. It is not now necessary to delve into this subject further; however, the student should remember that, for source material, the governing regulation is to be found in Part 40, and that a minimum quantity is involved.

The regulations governing special nuclear materials (SNM) are more extensive because of their weapon potential. The pertinent governing regulations may be found in 10CFR70 through 10CFR73.

In dealing with SNM, there is no exempt quantity - a license is required in every case. Part 70 covers licens-

ing and, not very surprisingly, exempts the Department of Defense (DOD) from the need to have a license. The DOD, because of its defense responsibility, maintains our nuclear weapon inventory.

Part 71, now defunct, dealt with packaging of all types of radioactive materials, including SNM; Part 73 deals with physical protection required for SNM in transit. The criticality aspects of shipping SNM included in Part 73 limit package content (10CFR71.6) so that, even if a package were to fall into water, it would not go critical.

Shipping of spent reactor fuel is handled as an exemption; here package design must undergo thorough analysis.

In addition to regulations found in the 10CFR70's, the Department of Transportation (DOT) has regulations covering radioactive shipments. These can be found in 49CFR, Chapter I.

Section 10CFR73.124 stipulates monitoring requirements in areas where SNM are stored or used. Compare facility monitoring systems with this regulation to ensure that they comply.

There are, as indicated before, many rules for byproduct materials. There are exemptions for small quantities (10 CFR30.70), and a license is required for all quantities in excess of those values. In addition, waste disposal regulations of Part 20, particularly of 10CFR20.303, which deals with disposal into sanitary sewage systems, must be considered.

Limits for radioactive material concentrations in air and water can be found in Appendix B of Part 20. Appendix B contains two tables: Table I gives concentration limits for restricted areas; Table II gives limits for unrestricted areas. The definition of a restricted area is one over which the facility can maintain access controls; an unrestricted

area is the entire remainder of the world. It is assumed that only employees can enter the restricted area and that they are there 40 hours per week. In the unrestricted area, occupancy is assumed to be 168 hours/week (7 x 24).

Each table is divided into two similar columns: Column I for air; Column II for water. So, in the restricted area, for each radionuclide, if the existing concentrations are at or below the concentrations indicated in the columns, people can drink or breathe safely. The same is true in the unrestricted area, except the applicable numbers are obviously found in Table II.

Actual values are referred to in the trade as "maximum permissible concentration" or, more commonly, MPC. In actuality, concentrations may be less than, equal to, or more than MPC. If things are under control, concentration in an unrestricted area will never exceed MPC. If concentrations in the unrestricted area do exceed MPC, through the fault of someone at the facility, this constitutes an uncontrolled release to the environment, which - as a minimum - requires many reports and - as a maximum - a stiff fine (minimum $5,000) and possible loss of facility license.

In a restricted area, the situation is different, as described in 10CFR20.103. Water activities in excess of MPC are usually no problem, because no one who is knowledgeable of this concentration will drink such water. Air activity in excess of MPC can be handled in two ways. First, one can reduce the time that such air is breathed and second, one can use respiratory protection equipment which has been shown to comply with the requirements of Part 20.103.

For example, suppose an airborne activity area exists in which the activity is determined to be 2,000 times MPC.

The best possible respiratory protection available reduces this by a factor of 1,000; thus, allowable work time must be reduced from 40 hours per week to 20 hours per week. Conversely, if the level is less than MPC, allowable work time can be lengthened. Remember, though, this refers only to a restricted area.

What should be done if more than one radionuclide is contributing to water or air activity? That situation is covered at the end of 10CFR20, Appendix B. This rule states that, if the concentration of each nuclide is divided by its respective MPC, the total of all such ratios must be less than or equal to 1.

For example, if the sum of the ratios is 3, then the composite airborne activity is three times MPC. MPC has thus been exceeded! To exhibit how this works, assume air activity composed of argon-41 and nitrogen-13. If the concentration of each, divided by its respective MPC, is 1/2, it is permissible (or, if the ratios come out 3/4 and 1/4, it is permissible). But if argon-41, say, is at the 1 MPC level, then any nitrogen-13 will cause the sum to exceed the total allowable MPC.

With unknown mixtures, or what are more correctly called "partially identified mixtures," releases to the environment are regulated by note 3 (b) at the end of Appendix B and, in the case of some facilities, by the Environmental Technical Specifications (Tech Specs). These specifications, as indicated earlier, are part of the facility license. The Tech Specs are usually more restrictive than 10CFR20, so are definitely a consideration in complying with regulations.

The terms "compliance" and "complying" have a definite meaning here, because facility records are periodically

audited by an inspector from the NRC's Division of Compliance. These auditors are knowledgeable people who have a responsibility to initiate various forms of citation procedures for noncompliance. In general, facility management does not like to be involved in citation proceedings.

In addition, depending on applicable local and state laws, there may be periodic visits from state officials, particularly for an x-ray facility. Such a facility is not under federal regulations included in Title 10, but is subject to Title 21, Chapter I, which deals with the Food and Drug Administration. X-ray equipment regulation is frequently carried out by the individual states, and pertinent state regulatory information can be obtained by first locating the proper state agency and then contacting them for applicable regulations.

In addition, there are so-called "agreement states." As of early 1975, 21 such states (primarily in the South, Southwest and Western portions of the country) came under the regulations governing agreement states (10CFR150). This setup gives the states power to regulate byproduct, source and SNM in quantities too small to constitute a critical mass. (If quantity of SNM is above limits in 10CFR150, federal control applies.)

In actuality, the federal government does retain control, because it licenses the agreement states to perform these regulating functions. These licenses can be rescinded at the government's discretion.

If someone is working in radiation protection in an agreement state, compliance with the state regulations is a must, and an audit by the state government can be expected. Some states, in spite of federal pre-emption of regulatory functions, have a complete set of regulations, encompassing

far more than 10CFR150. These state regulations are not necessarily legally binding because of federal pre-emption in nuclear matters, but it is normally convenient to comply with them, and most facilities go along with them to avoid a confrontation (as long as they do not hinder facility operation).

Federal regulation 10CFR19 deals with indoctrination and information that must be given to individuals working in radiation/contamination areas. It was originally intended to cover the indoctrination of construction workers who came into a facility to do a task that might require encountering radiation or contamination. It still covers that, but it also covers necessary instruction in radiation protection to be given every newly hired employee before that employee is permitted to move around in a facility alone.

One other source of regulations that may apply to a given facility is the union contract. At some facilities, local labor-management teams have agreed to certain work rules, which are usually more procedural than regulatory. For example, under certain circumstances, the union may require that a radiation monitor continuously check radiation levels during a job to more adequately protect workers. This rule has the advantage of allowing workers to proceed without worrying about radiation and may, in fact, turn out to be a faster way to get the job done and minimize exposure at the same time.

In summary, many regulations at the federal and state levels pertain to various aspects of radiation protection. The student should become familiar with the location and general content of federal regulations because they are applicable in all states. And, at each facility, a new

employee should study the pertinent state, company and union regulations to determine under what conditions they will be more restrictive or require additional reports.

Remember, these regulations are for the employee's own welfare!

EXERCISES

1. Worker X is employed as a radiation monitor at the local nuclear power plant; X is 30 years of age and received 25 rem of ionizing radiation at her last job as a nuclear fuel inspector; assume no other past radiation received. What is X's allowed ADE?
   5(N-18) - radiation received at past employment.
2. After determining X's allowed ADE, calculate limit on radiation she can receive during next 12 weeks.
   Hint: Don't forget 10CFR20.
3. Describe actions required as a result of each of the following incidents:
   - Exposure to 26 rem, whole body in one day from misuse of a radiography source. (10CFR20.403.)
   - An uncontrolled release to the environment, which exceeds 500 MPC when averaged over a 24-hour period. (10CFR20.403.)
   - Facility evacuation, which causes cessation of normal operations for four days. (10CFR20.403.)
   - Non-nuclear associated hydrogen explosion in a hot lab, which causes about $4,000 in damages. (10CFR20.403.)

MATERIALS

None.

PROCEDURES

None.

DATA TABLE

None.

REFERENCE MATERIALS

U.S. Department of Health, Education and Welfare, Food and Drug Administration. Title 21, Code of Federal Regulations, Chapter 1.

U.S. Department of Transportation. Title 49, Code of Federal Regulations, Chapter 1.

U.S. Nuclear Regulatory Commission. Title 10 - Atomic Energy, Code of Federal Regulations, Part 20, "Standards for Protection Against Radiation."

Government documents available by writing the Superintendent of Documents, U.S. Government Printing Office, Washington, D.C.

TEST

Read each question carefully to be sure that you give the type of response each question requires. Your answers should be brief. Use the backs of the test if you need additional space.

1. What does the term "maximum accumulated dose equivalent" mean?

_____

_____

2. To what document would you refer in order to find the accumulated dose equivalent? (Give the exact title and part).

_____

3. What information must be provided on an Nuclear Regulatory Commission (NRC) 4 form?

_____

_____

4. Write the quarterly maximum permissible dose equivalents for the following:

	Body Area	Limit Per Quarter
a.	whole body	_____
b.	extremities	_____
c.	skin	_____

5. Given a radiation worker's age, Accumulated Dose Equivalent (ADE) and the radiation level in the work area, complete the table below by calculating the following: (1) maximum quarterly exposure allowance; (2) amount of working time (in units of 40-hour work weeks) possible without exceeding the quarterly allowance.

Worker's Age	ADE	Radiation Level	Quarterly Exposure Allowance	40-Hour Work Weeks
19	5 rem	2 rem/hr.	_____	_____
30	15 rem	200 mrem/hr.	_____	_____
20	0.5 rem	100 mrem/hr.	_____	_____
45	2.5 rem	50 mrem/hr.	_____	_____

6. Calculate the Accumulated Dose Equivalent (ADE) for personnel with the following histories:

Age	Lifetime ADE Received	ADE Bank
19	0 rem	_____
22	0 rem	_____
22	3.5 rem	_____
50	21 rem	_____
28	50 rem	_____

7. Answer these questions about source materials:

   a. Where (in what document) would you find regulations that govern source materials? (Include the specific parts)

   _____

   b. How are source materials defined in the government document that pretains to source material?

   _____

8. Answer these questions about special nuclear materials:

   a. What, if any, group or organization is exempted from needing a license to work with special nuclear materials?

   _____

   b. What document contains regulations pertinent to special nuclear materials. (Include the specific part(s).)

   _____

   c. Would such things as luminous watch dials and radiographs be covered by regulations pertaining to special nuclear materials?

   _____

9. What are agreement states?

   _____

   _____

10. Write "yes" in front of all examples that are governed by Title 10, Code of Federal Regulations. Write "no" in front of all examples that are not governed by 10CFR.

a. _____ Limits for radioactive material concentrations in air and water

b. _____ Releases to the environment from partially identified mixtures

c. _____ The buying, selling, or use of byproduct materials

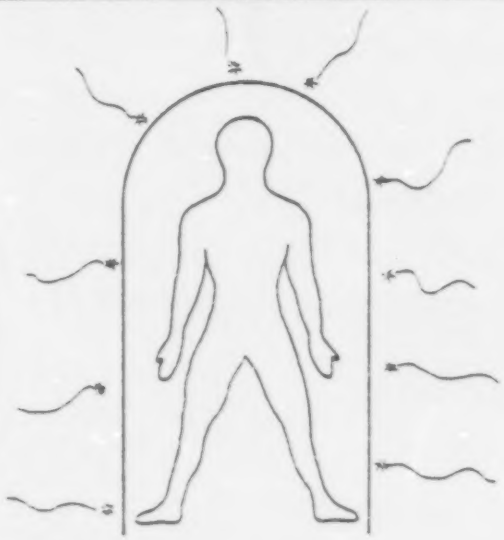
d. _____ X-ray facilities

e. _____ The indoctrination and information given to personnel working in radiation/contamination areas

f. _____ The range and accuracy of personnel dosimetry

NUCLEAR TECHNOLOGY

MODULE 2-4 INTERNAL DOSE CALCULATIONS



RADIATION PROTECTION I

COURSE 2

COURSE MODULES

COURSE 2 RADIATION PROTECTION I

MODULE 2-1 Radiation Exposure and Absorbed Dose Values

MODULE 2-2 Biological Effects of Radiation

MODULE 2-3 Radiation Protection Standards

MODULE 2-4 Internal Dose Calculations

MODULE 2-5 Personnel Monitoring Devices, Ionization Types

MODULE 2-6 Personnel Monitoring Devices, Chemical Types

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INTRODUCTION

Radiation exposure may result from external radiation (radiation sources outside the body) or internal radiation (radiation sources inside the body). Exposure from external radiation, either beta, x, gamma or neutron, can be easily calculated from mathematical expressions, or estimated from personnel monitoring devices, such as pocket dosimeters.

Exposure from internal radiation, usually either beta or alpha particles, is much more difficult to determine because 1) the mathematical expressions are complex and depend on many factors (which, in many cases, must be estimated) and 2) personnel monitoring devices do not measure internal radiation as such.

In addition, internal radiation has the potential to produce greater effects than external radiation because 1) it produces continuous exposure until it decays or is purged from the body by excretion and 2) the energies and specific ionizations for alpha particles, which are not considered an external hazard, can have a significant effect when the radionuclide is deposited directly in internal tissue.

This module gives the student an understanding of how an internal radiation dose is determined. Even though a technician would never perform an entire dose assessment, it is important to understand how the calculation is performed and how the technician may interface with the health physicist. This module also provides a background for determining surveys, smears and bioassays - tasks a radiation protection technician is expected to perform on the job.

MODULE PREREQUISITES

The student should have a good understanding of basic radiation terms and have completed Module 2-2, "Biological Effects of Radiation."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Describe three ways that radioactive materials can enter the body.
2. Describe the physical forms of radioactive materials that appear as internal radiation.
3. Differentiate between internal and external radiation sources, from a biological damage standpoint.
4. Describe the following techniques (used to determine if internal exposure has occurred):
   - Bioassay.
   - Whole-body count.

   (Include, for each technique, the specific purpose, probable frequency of usage, equipment used and general usefulness of results.)
5. Given the radioactivity from a bioassay sample, determine whole-body dose.
6. Given a whole-body dose, organ dose, total body burden (or organ burden), state whether or not recommended ICRP radiation protection guides are exceeded.

7. Define or describe the following terms (as discussed in this module):

Smear.
Critical organ.
Feces.
Radiosensitivity.
Quality factor (QF).
Effective half-life.
Ash sample.
Bioassay.
Body burden.
Aliquot
Chronic intake.
Acute intake.
Empirical equation.
Standard man.
MPC.
Handbook 69.

OVERVIEW OF INTERNAL RADIATION DOSE ASSESSMENT

Internal radiation dose assessment usually begins, quite naturally, by determining that an internal radiation source is present. In many cases, routine environmental laboratory surveys, such as air sampling and wipe tests, may indicate the need to perform additional tests, such as bioassays or whole-body counts, to identify a possible internal radiation hazard. Another frequent indicator is visual identification, such as noticing a leaking liquid source or a spill on the floor. In these latter cases, the radionuclide causing the contamination is usually known from the type of facility and material(s) being handled.

Once an internal hazard is determined, actual individual exposures can be estimated and compared with radiation protection guides to assess the extent of any possible biological hazard.

This module first discusses sources of internal hazards, how they enter the body, and what factors affect the extent of these hazards. Second, sampling methods are discussed; e.g., instrumental surveys and smear methods used to determine the amount of contamination. Third, methods to determine internal body exposure relative to radiation protection guides from the bioassay results are discussed.

INTERNAL HAZARDS

Radioactive materials may gain access into the body by ingestion, inhalation, absorption through the skin or by injection. The danger of ingesting radioactive materials is not necessarily that of swallowing a large amount at one time but, rather, the accumulation of small amounts over a protracted period of time.

INHALATION

Inhalation of radioactive aerosols is one of the principal potential modes of entry into the body. The lung, or primary receiver, is considered from two points of view: As the entry point into the body, and as a critical organ for retained radioactivity. The absorption, retention and elimination of material taken into the lungs depends upon such parameters as particle size of the inhaled material, solubility and rate of respiration of the individual. This presents a very complex situation.

INGESTION

The amount of absorption into the bloodstream of a radionuclide that enters the body through the mouth principally depends on its chemical and physical form. A large proportion of ingested material may be fairly rapidly excreted in the feces; but, because of irradiation of the intestine during the elimination process and because even a small amount entering the bloodstream is highly toxic, ingested radioactive material is very hazardous.

THROUGH SKIN

Radioactive materials can also enter the body by absorption through the unbroken skin or through abrasions, cuts, punctures and so on. Thus, all personnel working with radionuclides employing proper procedures should wear protective clothing to prevent material contact with exposed skin.

SOURCES OF INTERNAL HAZARDS

Any radioactive material that gains entry into the body is an internal hazard. The extent of the hazard depends upon the type of radiation emitted, its energy, the physical and biological half-life of the material and the radiosensitivity of the organ where the radionuclide localizes, as well as the quantity of the radionuclide. Alpha and beta emitters are the most dangerous radionuclides from an internal point of view, because their specific ionization is very high.

Alpha Emitters

As discussed earlier, because of their relatively large mass and double positive charge, alpha particles lose their energy very rapidly in any medium through which they pass; that is, the stopping range of alpha particles is very short. A 4-MeV alpha particle has a range in biological tissue of only a few hundredths of a millimeter. It can travel only 3 centimeters in air. Accordingly, alpha particles do not usually present an external radiation hazard, but do present one of the greatest possible internal hazards. When deposited throughout a vital organ, they can cause great damage because of their relatively high energies (4 to 9

MeV) and high specific ionization, and because their relative damage factor to tissue is about ten times greater than that for beta and gamma radiation.

Among the alpha-emitting nuclides in common use are plutonium-239, polonium-210, natural and enriched uranium and radium-226.

Beta Particles

Beta particles are lighter and have less charge than alpha particles. They, therefore have a longer range of travel and less specific ionization.

Accordingly, the quality factor (QF) of beta particles is much less than that for alpha radiation. For example, the absorbed total energy from beta radiation must be approximately ten times greater (per unit mass of tissue) than that of alpha radiation to produce the same damaging biological effect.

Both alpha- and beta-emitting materials are considered to be primarily internal radiation hazards; however, beta radiation may also present an external hazard when the energy is sufficient to penetrate the dead outer layer of skin. In fact, external beta burns can be extremely painful and slow to heal.

Among the common beta-emitting isotopes are strontium-90, yttrium-90 and iodine-131. Strontium-90, a fission product, is a pure beta emitter that decays with a half-life of 27.7 years. Its radioactive daughter, yttrium-90, is also a pure beta emitter. Strontium-90 has an effective half-life in the bone of approximately 17.5 years. Iodine-131 is also a fission product and a beta-gamma emitter.

FACTORS THAT AFFECT EXPOSURE AND INJURY

Many factors affect the amount of exposure from an internal deposition of radioactive material, as well as resulting biological damage. It is essential that a health physicist know the effect of each of these factors so that the potential for internal exposure injury can be properly assessed. These several factors are generally not completely understood or available, however. In most cases, assumptions must be made to relate an environmental concentration to which a person is exposed to the amount of the radionuclide received.

For example, consider a radiation worker who has breathed air containing a concentration of 2×10^{-9} $\mu Ci/cm^3$ of ^{131}I for a 40-hour week. The internal radiation dose could be calculated as a certain number of rads, based upon the amount of material inhaled, breathing rate and so on. This dose would not be very precise, but it would provide a useful guide for evaluation of the particular incident.

BIOLOGICAL RESPONSE

Several important factors that affect internal biological response require brief discussion.

METHOD OF ENTRY

The method of entry is of importance in determining

possible biological results following an exposure. Thus, a radionuclide entering via the respiratory tract may produce a detrimental result in the lung (and in the gastro-intestinal system, because some material is expelled from the respiratory system). If ingested, it will have its effect concentrated in the gastro-intestinal tract.

Thus, the mode of entry into the body becomes important. And, for each radionuclide, a statement of the type and degree of injury following each mode of entry into the body must be assessed to allow evaluation of all possible consequences.

QUANTITY AND TYPE OF RADIONUCLIDE

Exposure depends on environmental concentration and length of time exposed to that concentration. Exposure also depends on type and energy of radiation. Remember, hazard is proportional to quality factor of radiation; e.g., alpha radiation is more damaging than beta radiation for a given energy. As energy of radiation is increased, damage increases (usually).

INITIAL BODY RETENTION

Large fractions of some elements are absorbed strongly when taken into the body, while others exhibit less retention. The hazard, obviously, depends on the amount of material retained by the body. The amounts not absorbed are rapidly removed by elimination through the gastro-intestinal tract, exhaled through the lungs, and so on. In addition, different materials are concentrated or retained

in different organs. That is, there may be some specificity in location.

RATE OF ELIMINATION

The rate of elimination from blood and tissues of the body varies considerably for different elements or compounds. The time required for one-half of the original quantity of radioactive material to be removed from the body by biological processes is called "biological half-life."

By assuming that biological elimination is exponential, radioactivity half-life and biological half-life may be combined into a single effective half-life by means of the formula:

$$T_{eff} = \frac{(T_p) \times (T_b)}{(T_p) + (T_b)}$$

Equation 2-4-1
(Equation = Eq.)

where

T_{eff} = effective half-life.

T_p = radioactivity half-life.

T_b = biological half-life.

EXAMPLE

Given: ^{131}I is a beta-gamma emmitter with an average gamma energy of 0.37 MeV. Its physical

half-life is 8 days, with a biological half-life of 140 days.

Find: Effective half-life:

$$T_{eff} = \frac{8 \times 140}{8 + 140} = 7.6 \text{ days.}$$

The shorter the effective half-life of a radionuclide, the lower the potential for damaging effects.

Some materials are eliminated rapidly from the body, whereas large fractions of others remain in specific body organs. For example, radium, plutonium and strontium deposit in bones, where the elimination rate is very slow; i.e., the biological half-life is many years. Remember that all isotopes of an element behave chemically the same and, thus, have the same biological half-life, even though the physical half-lives are different. Radioisotopes of the above elements are much more hazardous than those of carbon, sodium and sulfur, which have biological half-lives ranging from a few days to a few weeks.

The principal biological methods of elimination of radionuclides from the body are urine, feces, exhalation and perspiration. Usually, elimination is fairly rapid for those radionuclides that remain in the gastro-intestinal tract. For those radionuclides that are absorbed into the bloodstream and transported to a permanent site, such as bone, elimination is much slower. After the initial period,

the elimination rate becomes more nearly exponential, and the application of the term, biological half-life has more significance.

RADIOSENSITIVITY

Radiosensitivity refers to the relative susceptibility of body organs and tissues to radiation damage. Certain body organs or tissues are more radiosensitive than others. For example, the lymphatic tissue and bone marrow are much more radiosensitive than muscle or nerve tissue; therefore, in equal concentrations, an element like plutonium is more hazardous than uranium because plutonium concentrates in the most sensitive part of the bone (marrow) and is retained for long periods. In contrast, uranium goes to other portions of the bone, to the kidneys or to other relatively insensitive organs. Also, it is eliminated faster from the body.

ORGAN SIZE

For a given number of microcuries of a radionuclide in a critical organ, the smaller the organ, the greater the concentration and the greater the dose delivered to the critical tissue. Iodine presents a much greater problem than sodium: iodine is very selectively absorbed in a much smaller organ, the thyroid gland; sodium is rather uniformly distributed throughout the whole body. In many cases, the radionuclide is deposited in a large organ but becomes localized in a small portion of that organ, so that - in effect - the mass of the critical tissue there, too, may become very small.

ORGAN ESSENTIALITY

Some organs are either not essential to the body function, or, when damaged or removed, special steps can be taken to supplement or compensate for their reduced functional capacity. For these reasons, damage in some cases to the bone marrow, kidneys, eyes, etc., may represent a greater hazard than equal tissue damage to some gland such as the thyroid.

In summary, the biological effects of internal emitters are the same as for external emitters; that is, the mechanisms for damage to the cell and tissues are the same. (Specific biological effects were studied in an earlier module.) But internal emitters are usually much more hazardous than external emitters because:

They produce continuous radiation until they decay and/or are excreted from the body.

They may tend to localize in certain organs as a result of body chemistry.

PREVENTION OF INTERNAL HAZARDS

The prevention and control of airborne contamination is the most effective way of reducing internal hazards. The use of protective devices and the employment of proper procedures and good handling techniques are instrumental in this respect.

Laboratory operations with radioactive materials should be carried out in hoods so designed that room air contamina-

tion is kept at a minimum. Exhaust air must be filtered, and, if necessary, washed to eliminate any possible public hazard. Protective clothing should be worn at all times so that permanent clothing does not become contaminated. This measure helps to limit spread of contamination.

To prevent inhalation of radioactive materials, respirators should be available for emergency operations in any area where a concentration of airborne activity is above maximum permissible levels.

Eating and smoking in areas where radioactive materials are handled should be prohibited to reduce the ingestion hazard.

Laboratory design and construction should be such that, if decontamination becomes necessary, it is easily accomplished. It may be desirable to coat floors, walls, ceilings and furniture with strippable material so contamination may be removed and disposed of readily.

SAMPLING METHODS

AIR MONITORING

The purpose of an atmospheric sample measurement is to determine as accurately as possible the original concentration of radioactive material per unit volume of the sampled air. The radioactive material may be measured in several ways:

On the sampling medium.
On material removed from filter.
On the entire filter.

To facilitate measurement, a sample may be concentrated by dissolving it in an appropriate solvent or by taking an ash sample of the medium in an oven or furnace. The specific method of measurement depends on the particular radioactive material being investigated.

In the case of alpha-emitting materials (e.g., uranium or thorium), care should be taken to avoid self-absorption. Alpha particles emitted from materials embedded in a filter or in a sample of ash may be absorbed in the filter or ash and not be detected. This absorption may also occur to a smaller extent in low-energy beta-emitting materials.

For alpha measurements, scintillation methods and alpha proportional counters are most frequently used. In some cases, the radioactive sample is placed within a proportional counter as in a gas-flow proportional counter. In other cases, to attain uniform distribution of the radioactive sample, it may be placed on a sample holder or planchette.

Sampling and measuring of radioactive gases presents particular problems. Some of the elementary gases that must sometimes be sampled include tritium, argon, krypton, neon, radon and xenon. A gas may be concentrated by collecting it through absorption on activated charcoal. It then may be measured on the charcoal, or removed from the charcoal by heating.

Many common gases, such as sulfur dioxide and carbon dioxide, that have radioactive sulfur or carbon, respectively, must be sampled frequently. The usual methods of absorbing nonradioactive gases are also utilized to concentrate those with radioactivity. They may then be measured in the conventional method. Although iodine-131 is of considerable interest, it usually does not appear as a gas, but rather as an iodine salt, or in a vapor, or absorbed on dust particles.

WATER MONITORING

For determining beta-gamma activity in water, it is possible to pass water under detectors and continuously record its radiation level. For alpha emitters, however, up to the present time, such external detection has not been found practical, although efforts are still being made to use certain types of scintillation counters for this purpose. In any event, what are probably the most precise determinations of the concentrations of a radioisotope in water are made by the normal radioanalytical techniques of evaporation, concentration of the radioisotope and subsequent counting of the residues.

SURFACE MONITORING

As mentioned, exposures from the environment, particularly for alpha-emitting materials, lie in the possibility of such materials being released from work surfaces or brushed off clothing and inhaled. In general, normal radiation detection methods (including the types of instruments mentioned previously) can be used to monitor clothing and surfaces. In some cases, it has been found desirable to use special types and shapes of probes.

A "smear test," wiping a piece of filter paper lightly over a suspected area, is used to determine if the radioactive material is loose or fixed; loose material, which adheres to the smear paper, can then be counted in a laboratory.

It should be re-emphasized that all environmental monitoring of air, water and, particularly, surfaces, reflects only second- and third-order possibilities of actual internal

radiation exposures: material must actually enter the body to be a problem. This statement is true, even for beta-gamma emitters. Environmental surveys of beta-gamma fields can indicate a direct potential exposure in the sense that an individual's body is in the same radiation field that the individual's survey meter encounters.

BIOASSAY METHODS

Monitoring methods mentioned previously are used to determine radionuclide concentrations in the environment. The results of these methods are then used indirectly to determine how much of a radionuclide entered the body. Bioassay methods present a direct method of determining internal radiation exposure.

Estimation of how much of a radionuclide is present in the body can be done by one or more of several methods, usually depending on the radionuclide and its mode of excretion. For example, the radium content of the body may be estimated by measuring the radon content of breath samples, or by means of a whole-body gamma scintillation technique, or by measuring the alpha activity of radium in urine or feces.

The method selected for a bioassay of any particular radionuclide is primarily determined by the ease, available sensitivity and reproducibility of the method. In bioassays, the following factors are important.

Type of Sample

Bioassay procedures essentially consist of excretion analysis (e.g., urine, feces or breath) or whole-body count-

ing (by means of gamma scintillation techniques). Bioassay of nasal smears, sputum and blood are only rarely done.

In excretion analysis, the most widely used assay medium is urine, although feces analysis theoretically can give the same measure of the body burden of a radionuclide. Except for the special cases of radium and carbon-14, breath analysis is not usually done.

Immediately after internal exposure to a known or suspected radionuclide, several successive 24-hour urine specimens and perhaps several feces specimens should be obtained, depending on the specific radionuclide. Nasal smears and sputum samples are also sometimes very helpful, at this point, to establish the fact of actual exposure and to help identify the radionuclide.

It is theoretically possible to determine the acute absorbed dose after taking a single urine specimen or feces specimen at any time after exposure. Such a specific determination can only be done if the excretion rate of the radionuclide, as a function of time, is known beforehand.

In collecting samples for bioassay (especially urine specimens), it is essential that they be obtained with the utmost care to avoid further contamination.

Types of Exposure

Acute inhalation or one-time ingestion of a radionuclide causes the level of radioactivity present in the urine and/or feces to rise very quickly, perhaps within hours. A peak is reached, usually within a day, and then activity falls off, at first rapidly, then more slowly. Activity may still be noted in some instances after months or years.

With chronic ingestion or inhalation of a radionuclide, the radioactivity in the excretion samples is first noted

to increase constantly. Finally, a point is reached beyond which no further rise in activity is noted. At this point, an equilibrium has been reached and the body intake equals the body excretion.

Specific Types of Excretion Analysis

Alpha Activity. In cases where there is a possibility of ingestion or inhalation of a number of different alpha emitters, a gross alpha analysis of urine samples indicates the overall amount of radioactive material in the body.

A procedure employed by one commercial firm is to extract the alpha emitters from the urine after ashing and counting. If a significant amount of radioactivity appears to be present, specific radionuclide assays are then performed.

Beta Activity. Analysis of urine for gross beta activity is commonly used to indicate inhalation or ingestion of fission products or other beta emitters. Since potassium-40 is normally present in urine, it is necessary either to measure total potassium content and, from its isotopic abundance, determine potassium-40 contribution to the count rate, or to chemically remove potassium before counting.

Either of two methods can be used for determining net beta activity of urine. One involves ashing of the sample, dissolving the ash and diluting to a known volume, evaporating an aliquot (fraction of a total sample) for beta counting, and taking another aliquot for flame photometry measurement of total potassium. The potassium-40 contribution is then calculated from its natural isotopic abundance and subtracted from the gross beta count to give a net count.

Analysis for Specific Radionuclides

In most bioassays, only one or two radionuclides are of concern (because of the facility type or radionuclide generally used); therefore, specific techniques can be used to separate a specific chemical from a sample. Each technique should include the type of sample (e.g., urine), frequency of sample collection, separation method and counting method.

Whole-Body Counter

Another bioassay technique, in which no excretion sample is necessary, is the whole-body counter. Frequently, this method determines the gamma-emitting elements in vivo (in the living body) by a NaI scintillation system and pulse-height analyzer. With this apparatus, some identification of the radionuclides can be made within a very short time. Because most radionuclides emit gammas along with alpha or beta radiation, the results can be used to determine alpha-beta activity.

INTERNAL DOSE CALCULATION

In the previous section, methods of obtaining samples were described. From the samples, it is possible to determine the amount of radionuclides deposited in vivo and their distribution within the body. Radionuclide determination in vivo is a very complex problem and will not be discussed at this time.

Assessment of the degree to which exposure has occurred

as a result of these radionuclides is also a very complex problem; the method or methods chosen for exposure assessment may range from application of simple clinical laboratory techniques to highly sophisticated techniques involving advanced instrumental analysis. The overall objective is still the same, to evaluate the degree of radiation exposure.

In this section, a general simplified explanation of the process used to determine exposure will be given. Recall that most radionuclides do not distribute themselves uniformly in the body, but concentrate in certain organs and, because certain organs are more radiosensitive than others, some organs receive more damage than others.

Calculations are generally based on the organ of the body whose damage by the radiation results in the greatest damage to the body. This organ is defined as the "critical organ." One of the best known examples is the thyroid gland, which is the critical organ for iodine-131.

CHRONIC VERSUS ACUTE INTAKE

The discussion herein deals with two types of situations, chronic intake and acute intake. Chronic intake means the radionuclide is being continually taken into the body over a long period of time. An example of chronic intake is the ingestion over a long period of time of milk or water that contains strontium-90 or other fission products. An acute intake refers to ingestion or inhalation of a quantity of some radionuclide over a relatively short period of time. The administration of a single diagnostic or therapeutic dose of iodine-131 would fall into this category. Calculations in this module only consider chronic intake.

SAMPLE DOSE-RATE CALCULATION FOR
CONTINUOUS (CHRONIC) INTAKE

Consider the dose rate to a specific organ at any time (t) after continuous intake of a single beta-emitting radionuclide has begun. The dose rate (D_r) is found by use of the following empirical equation (an equation based on experiment and experience)*:

$$D_r = 3.58 \times 10^2 \left(\frac{p}{m\lambda_{eff}}\right) E \left(1-e^{-\lambda_{eff}t}\right) \qquad \text{Eq. 2-4-2}$$

where

D_r = dose rate, rem/wk.

p = rate of uptake by organ, Ci per unit time.

m = mass of organ, gram.

λ_{eff} = effective body decay constant.

E = average energy per disintegration.

t = time after beginning intake.

e = constant (2.718).

Mass of the organ in question is determined from data on the standard man as given in the Radiological Health Handbook. Standard man represents a set of internationally agreed upon values for the many human body characteristics necessary for internal dose calculations.

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* The student will not be expected to use this equation, but only to recognize it, understand what it is to be used for and identify its terms.

The preceding equation enables calculation of an approximate biological dose to body organs resulting from radionuclides being deposited in the body. To say that the results are highly accurate in all cases would be a gross misrepresentation. As more knowledge is gained about the behavior of certain radionuclides in the body, and biological and physical constants, this understanding should change. For now, error may be introduced as a result of the assumptions stated previously. Even so, the relationships are extremely useful in establishing guidelines by which radiation exposure may be analyzed.

MAXIMUM PERMISSIBLE CONCENTRATIONS

From the internal dose calculations, relationships have been developed between exposure and concentration. From radiation protection guides (refer to Module 2-2), maximum permissible amounts of a given radionuclide in a body or an organ were established (called "maximum permissible body burden" and "maximum permissible organ burden"). The maximum permissible concentration (MPC) is that concentration that may be inhaled or ingested safely, either for radiation workers or for the general public.

By knowing the body burden or organ burden, it is possible to calculate the maximum permissible concentration of a radionuclide in air or water. Values for these concentrations are given in Handbook 69, "Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure."

A typical page from Handbook 69 is given in Table 2-4-1. Maximum permissible concentrations and body burdens are given for specific nuclides. Separate values are given for soluble and insoluble radionuclides because of the differences in absorptivity in the body organs.

SUMMARY

Internal dose assessment is employed for two prime purposes:

To determine maximum permissible concentrations of radionuclides in air and water, based on radiation exposure guides.

Once maximum permissible concentrations are known, it is simply a matter to keep concentrations below maximum (that is, to stay within the radiation exposure guides). Note: Maximum permissible concentrations are given in NBS Handbook 69 so that they do not have to be determined for every case in which the radionuclide is known. The flow chart (Figure 2-4-1) depicts the process used to determine maxi-

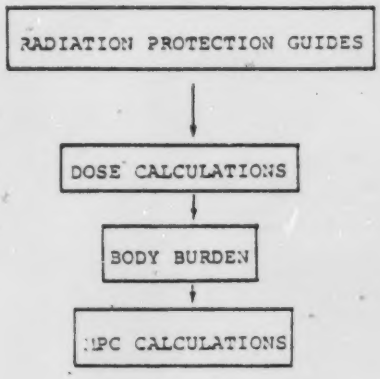

Figure 2-4-1.
Flow chart for establishing maximum permissible concentrations (MPC).

TABLE 2-4-1

MAXIMUM PERMISSIBLE BODY BURDENS AND MAXIMUM PERMISSIBLE CONCENTRATIONS OF RADIONUCLIDES IN AIR AND IN WATER FOR OCCUPATIONAL EXPOSURE

Radionuclide and type of decay		Organ of reference (critical organ in boldface)	Maximum permissible burden in total body μc	Maximum permissible concentrations			
				For 40 hour week		For 168 hour week**	
				(MPC)$_a$ $\mu c/cc$	(MPC)$_w$ $\mu c/cc$	(MPC)$_a$ $\mu c/cc$	(MPC)$_w$ $\mu c/cc$
$_{15}P^{32}$ (β^-)*		**Bone**	6	5×10^{-4}	7×10^{-6}	2×10^{-4}	2×10^{-6}
		Total Body	30	3×10^{-4}	4×10^{-7}	9×10^{-5}	10^{-7}
	(Sol)	**GI (LLI)**		3×10^{-4}	4×10^{-7}	9×10^{-5}	2×10^{-7}
		Liver	50	5×10^{-4}	6×10^{-7}	2×10^{-4}	2×10^{-7}
		Brain	300	0.02	3×10^{-6}	8×10^{-5}	10^{-6}
		Lung			8×10^{-6}		3×10^{-6}
	(Insol)	**GI (LLI)**		7×10^{-4}	10^{-7}	2×10^{-4}	4×10^{-6}
$_{16}S^{35}$ (β^-)		**Testis**	90	2×10^{-3}	3×10^{-7}	6×10^{-4}	9×10^{-6}
		Total Body	400	7×10^{-3}	10^{-6}	3×10^{-3}	4×10^{-7}
	(Sol)	**Bone**	800	0.02	2×10^{-6}	5×10^{-3}	8×10^{-7}
		Skin	3×10^3	0.07	10^{-6}	0.02	3×10^{-6}
		GI (LLI)		0.2	4×10^{-6}	0.05	10^{-6}
		Lung			3×10^{-7}		9×10^{-6}
	(Insol)	**GI (LLI)**		8×10^{-4}	10^{-6}	3×10^{-4}	5×10^{-7}
$_{17}Cl^{36}$(β^-)		**Total Body**	80	2×10^{-4}	4×10^{-7}	8×10^{-5}	10^{-7}
	(Sol)	**GI (LLI)**		0.04	5×10^{-6}	0.01	3×10^{-6}
		Lung			2×10^{-6}		8×10^{-6}
	(Insol)	**GI (LLI)**		2×10^{-4}	3×10^{-7}	6×10^{-5}	10^{-7}
$_{17}Cl^{36}$(β^-,γ)		**GI (S)**		0.01	3×10^{-6}	4×10^{-3}	5×10^{-7}
	(Sol)	**Total Body**	9	0.3	4×10^{-6}	0.1	2×10^{-6}
		GI (S)		0.01	2×10^{-6}	4×10^{-3}	7×10^{-7}
	(Insol)	**Lung**			10^{-6}		8×10^{-6}
$_{18}A^{37}$(e)	(Immersion)	**Skin**			6×10^{-3}		10^{-4}
$_{18}A^{41}$(β^-,γ)	(Immersion)	**Total Body**			2×10^{-6}		4×10^{-7}
$_{19}K^{42}$ (β^-, γ)		**GI (S)**		9×10^{-4}	2×10^{-6}	3×10^{-3}	7×10^{-7}
		Total Body	10	0.02	3×10^{-6}	8×10^{-3}	10^{-6}
		Brain	20	0.04	6×10^{-6}	0.01	2×10^{-6}
	(Sol)	**Spleen**	20	0.04	6×10^{-6}	0.01	2×10^{-6}
		Muscle	20	0.04	6×10^{-6}	0.02	2×10^{-6}
		Liver	50	0.08	10^{-6}	0.03	4×10^{-6}
		GI (LLI)		6×10^{-4}	10^{-7}	2×10^{-4}	4×10^{-6}
	(Insol)	**Lung**			9×10^{-7}		3×10^{-7}
$_{20}Ca^{45}$ (β^-)		**Bone**	30	3×10^{-4}	3×10^{-6}	9×10^{-4}	10^{-6}
	(Sol)	**Total Body**	200	2×10^{-3}	3×10^{-7}	7×10^{-4}	9×10^{-6}
		GI (LLI)		0.01	3×10^{-6}	4×10^{-3}	10^{-6}
		Lung			10^{-7}		4×10^{-6}
	(Insol)	**GI (LLI)**		5×10^{-4}	9×10^{-7}	2×10^{-4}	3×10^{-7}
$_{20}Ca^{47}$ (β^-, γ)		**Bone**	5	10^{-4}	2×10^{-7}	5×10^{-4}	6×10^{-6}
	(Sol)	**GI (LLI)**		2×10^{-4}	3×10^{-7}	8×10^{-4}	2×10^{-7}
		Total Body	10	4×10^{-4}	5×10^{-7}	2×10^{-3}	2×10^{-7}
	(Insol)	**GI (LLI)**		10^{-4}	2×10^{-7}	3×10^{-4}	6×10^{-6}
		Lung			2×10^{-7}		6×10^{-6}
$_{21}Sc^{46}$ (β^-, γ)		**GI (LLI)**		10^{-4}	2×10^{-7}	4×10^{-4}	3×10^{-6}
		Liver	10	6	2×10^{-7}	2	8×10^{-6}
	(Sol)	**Kidney**	10	6	3×10^{-7}	2	9×10^{-6}
		Total Body	20	6	3×10^{-7}	2	10^{-7}
		Bone	60	20	10^{-6}	8	4×10^{-7}

* An alternate and older form of identification, $_{15}P^{32}$ is equivalent to ^{32}P.

** The MPC values for 40-hour weeks are to be considered as basic for occupational exposure, and the values for 168-hour weeks are basic for continuous exposure, as in the case of the population at large.

mum permissible concentrations.

To determine internal exposure dose. This dose is added to the external dose to determine total exposure dose. Total dose, along with radiation protection guides, is used to evaluate biological damage. The flow chart (Figure 2-4-1) depicts the process used to assess internal dose (Figure 2-4-2).

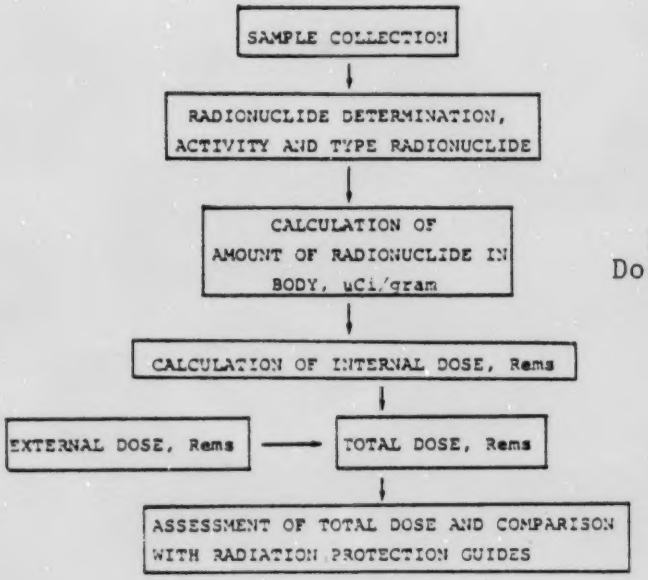
SAMPLE COLLECTION
RADIONUCLIDE DETERMINATION, ACTIVITY AND TYPE RADIONUCLIDE
CALCULATION OF AMOUNT OF RADIONUCLIDE IN BODY, uCi/gram
CALCULATION OF INTERNAL DOSE, Rems
EXTERNAL DOSE, Rems
TOTAL DOSE, Rems
ASSESSMENT OF TOTAL DOSE AND COMPARISON WITH RADIATION PROTECTION GUIDES

Figure 2-4-2.
Dose assessment.

The student should gain a sound overall understanding of the entire internal dose assessment problem to be able to better interface with the health physicist.

EXERCISES

An employee swallows a sample of radioactive substance that has a physical half-life of seven days. Under normal circumstances, one-half of this type of substance is eliminated from the body in two days.

Determine effective half-life of the substance.

MATERIALS

None.

PROCEDURES

Student should observe a complete internal dose assessment demonstrated either by the instructor or at an industrial radiochemistry laboratory, hospital, college or university. Student should record correct data in the spaces provided below, from personal observations:

1. Briefly describe one method by which each of the following samples was routinely collected, prepared and counted (be sure to list frequency of sampling, type of sampling equipment used and type of counting equipment used):

Air.

Water.

Surface.

2. For routine samples, describe how each was counted; give units of activity:

3. Describe bioassay technique used to determine internal radionuclide (include sampling, preparation and counting techniques); record resultant activity units and type of radionuclide of one sample:

4. List (on Data Table 2-4-1) radionuclides present (or possibly present); state ways by which each could enter the environment. For each nuclide, record half-life, type of radiation, MPC_w, MPC_a, body burden and critical organ:

5. Explain how health physicist took sample activity, and how it was related to a radionuclide concentration in the body:

6. Explain how health physicist took body (or organ) radionuclide concentration and how it was related to internal exposure dose:

7. Explain what the health physicist did with the internal absorbed dose or dose equivalent:

DATA TABLE

DATA TABLE 2-4-1

	RADIONUCLIDE	RADIOLOGICAL HALF-LIFE	BIOLOGICAL HALF-LIFE	EFFECTIVE HALF-LIFE	MPC_w	MPC_a	BODY BURDEN	CRITICAL ORGAN
1.								
2.								
3.								
4.								

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Kramer, Henry H. and Gemmill, Wayne J., eds. Experiments in Nucleonics. Tuxedo, NY: Union Carbide, Sterling Forest Research Center, 1968.

National Bureau of Standards (NBS). "Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure," in NBS Handbook 69. Washington, DC: United States Department of Commerce, 1959.

National Council on Radiation Protection and Measurements (NCRP). "Basic Radiation Protection Criteria," NCRP Report No. 39. Washington, DC: NCRP, 1974.

TEST

Read each question carefully. Be sure to provide the specific information required by the question. Your answers should be brief.

1. Give one reason internal radiation can produce greater biological effects than external radiation.

2. Why is radioactive material that is taken into the body through the mouth especially hazardous?

3. List two ways, besides ingestion, that radioactive materials can enter the body.

   a. _____

   b. _____

4. Why are alpha particles capable of causing great damage when deposited in a vital organ?

5. Compare the relative damage factor to tissue of alpha and beta particles.

6. What does the term "biological half-life mean?

7. Briefly describe how the following factors affect the amount of exposure from an internal deposit of radioactive material: initial body retention, radiosensitivity and organ size.

   Initial body retention:

   Radiosensitivity:

Organ Size:

_____

_____

8. What are the four principal methods by which radionuclides are eliminated from the body?

   a. _____

   b. _____

   c. _____

   d. _____

9. What is the purpose of the smear test? _____

   _____

   _____

10. In excretion analysis used to determine ingestion of a radionuclide, how would results differ for acute ingestion and chronic ingestion?

    _____

    _____

    _____

    _____

11. List in order the steps involved in evaluating biological damage due to radiation (You may instead of listing the steps, make a flow chart that shows the order in which the processes (steps) are undertaken).

12. Name two bioassay methods of determining internal radiation exposure.

    a. _____

    b. _____

13. Name one method for determining the net beta activity of urine.

_____

14. Given the following whole body doses, write "yes" in the space provided if the dose exceeds the recommended International Committee on Radiation Protection limits;(ICRP) write "no" if the dose does not exceed these limits.

	Whole Body Dose	Limit Exceeded (Yes/No)
a.	3 rem in 20 weeks, age 20	_____
b.	3 rem in 1 day, age 19	_____
c.	3 rem in 6 weeks, age 18	_____
d.	$^{42}_{19}K$, concentration 12 μc/cc per 40 hour work week	_____
e.	$^{32}_{15}P$, concentration 25 μc/cc per 40 hour work week	_____

15. Calculate the effective half-life for a beta-gamma emitter with a physical half-life of 10 days and a biological half-life of 180 days.

16. What does the term "standard man" mean? _____

_____

17. Where would you look to find the maximum permissible body burden and maximum permissible concentration of radionuclides in air and water?

_____

18. Suppose that a person ingested equal quantities, in Curies of two alpha-emitting radionuclides - one radionuclide having an effective half-life of 120 days, the other radionuclide having an effective half-life of 170 days. Would you expect any difference between the two nuclides in the amount of biological damage that could be caused? Why?

_____

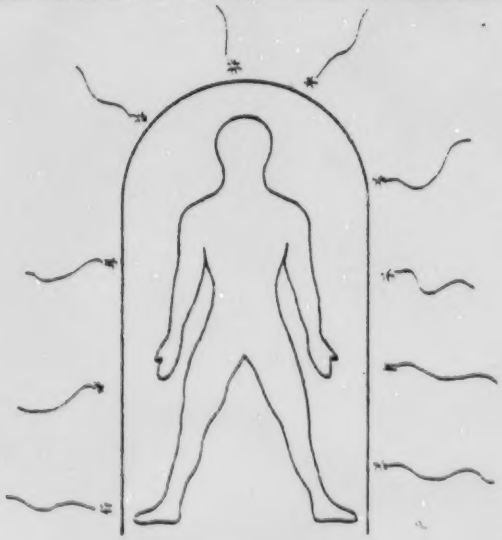
_____

_____

_____

NUCLEAR TECHNOLOGY

MODULE 2-5 PERSONNEL MONITORING DEVICES - IONIZATION TYPES

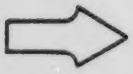


RADIATION PROTECTION I

COURSE 2

COURSE MODULES

COURSE 2 RADIATION PROTECTION I



MODULE 2-1	Radiation Exposure and Absorbed Dose Values
MODULE 2-2	Biological Effects of Radiation
MODULE 2-3	Radiation Protection Standards
MODULE 2-4	Internal Dose Calculations
MODULE 2-5	Personnel Monitoring Devices, Ionization Types
MODULE 2-6	Personnel Monitoring Devices, Chemical Types

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INTRODUCTION

The basic purpose of personnel monitoring devices (dosimeters) is to provide a rapid determination of individual radiation exposure. These devices are conservative: they indicate amount of radiation exposure larger than that actually being absorbed. Essentially, they tell immediately the amount of personal exposure - dosimeters do not have to be processed further, as do film or thermoluminescent badges.

Over the years, there have been many variations in dosimeter designs. For the most part, all have been based on the gas ionization principle, with variations generally in the type of readout. The simplest (and least expensive) type has no direct readout at all: it has to be taken to a special device to determine radiation dose. It suffers the disadvantage of not being a "right now" device.

This no-readout dosimeter is not the device to use if the special reading device cannot be conveniently situated near the work area. For example, in a radiation area where special clothing must be worn, entry and exit take considerable time, therefore a self-reading dosimeter is essential. For this application, the ideal device will have a visible or audible readout.

This module will concentrate on the two types of dosimeters most universally used in industry: the self-reader and the "pencil" type. For any other types encountered in practice, read manufacturer's operating instructions (principles of operation will be similar to those described here).

MODULE PREREQUISITES

A knowledge of basic source-handling safety precautions, as well as the procedure in Module 2-1, "Radiation Exposure and Absorbed Dose Values," for determining radiation levels at various distances from a point γ source, are required. A copy of Module 2-1 is necessary for carrying out the Procedures in this module.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Charge a dosimeter to a zero reading.
2. Read a dosimeter.
3. Explain operating principles of the two most common types of dosimeters (use sketches; indicate main parts).
4. Select proper type of dosimeter for a given set of expected radiation circumstances.
5. Enumerate three precautions and limitations to be observed in using dosimeters.

PRINCIPLES OF DOSIMETERS AND FAMILIAR OCCURRENCES

The dosimeters discussed here are approximately 5/8" in diameter and approximately one pocket-depth long (≈ 3 1/2"). Both work on the same principle as an electroscope. (Run a comb through your hair, then place comb near several small pieces of paper; likely the paper will become agitated because of an electrostatic charge difference between the comb and the paper. An electroscope (Figure 2-5-1) works in this way. If the comb is again run through the hair and touched to the charging rod of an electroscope, its two leaves will fly apart because both have the same charge, and like charges repel one another. If the electroscope is undisturbed, the charge leaks off and the leaves gradually fall back together in a straight, vertical position. The charge leaks to the case through the insulator, over a long period of time.)

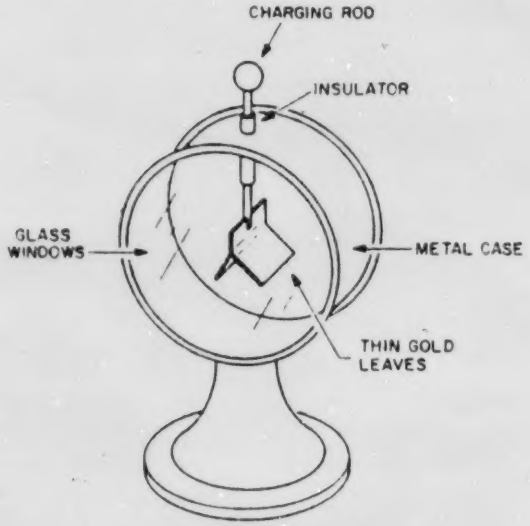


Figure 2-5-1.
The leaf electroscope (shown in charged condition).

If the positive terminal of a (few hundred volt) battery is connected to the knob of the electroscope, and the other terminal to the case, the leaves become "like charged" and fly apart, because the voltage moves electrons from the leaves to the case (Figure 2-5-2), leaving the leaves with

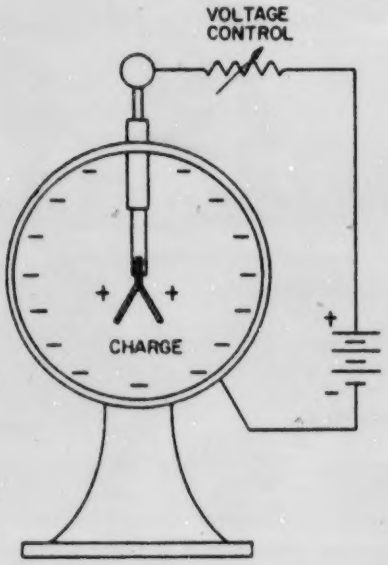


Figure 2-5-2.
Charging electroscope
with a battery.

a positive charge. If the battery is connected in reverse, the charge flows from case to leaves. Both leaves still have like polarity, and fly apart. By varying the voltage applied, displacement of the leaves can be accurately controlled. Such control is of no importance to an electroscope, but it is essential to dosimeter operation.

THE PENCIL DOSIMETER

The simplest ionization type dosimeter is the pocket

ionization chamber or, as it is commonly known, the pencil dosimeter (Figure 2-5-3). This device is simply an ioniza-

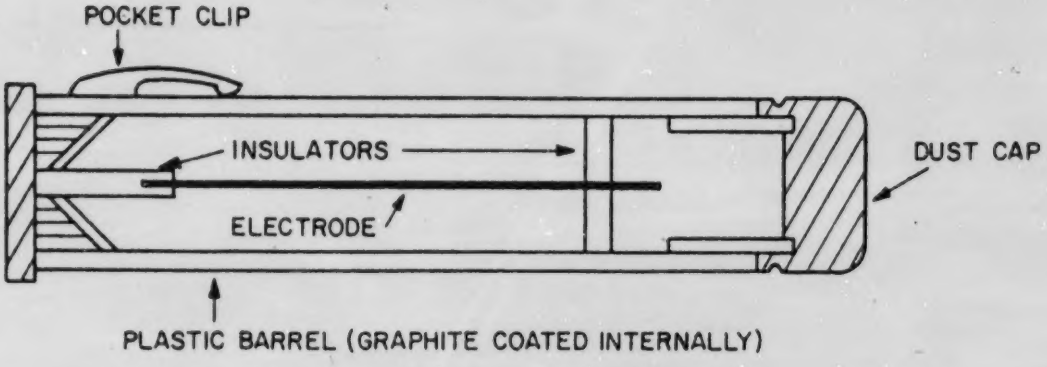


Figure 2-5-3. Pencil dosimeter.

tion chamber with a central electrode (electrically insulated from the case, which serves as the other electrode). A charge is applied to this device exactly as applied to the electroscope described above. Reading the chamber is accomplished with the help of an electronic circuit called a "charger-reader" (Figure 2-5-4), so named because it con-

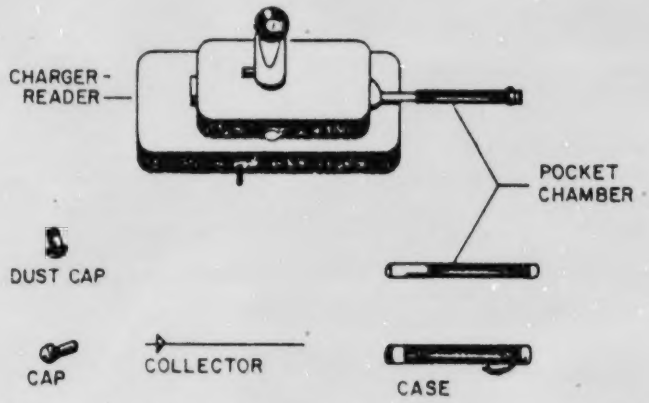


Figure 2-5-4. The dosimeter charger-reader.

tains the battery and control for charging the dosimeter as well as the circuit for reading it.

The maximum reading for this particular dosimeter (Figure 2-5-5) is 200 mR - the scale for the most common type - but other ranges are available, up to about 5 R.

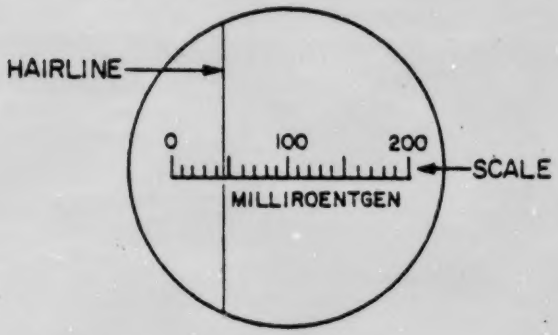


Figure 2-5-5. Eyepiece of a charger-reader.

The simplest charger-readers require only that the chamber dustcap be removed from the pencil and the cap end pushed into the reader to activate the read circuit. The more complicated types have off-on switches and read switches. In large facilities where many pencil dosimeters are read each day, reading is accomplished with a meter rather than an eyepiece.

Portability requirements and price usually dictate that a facility have several simple readers rather than one complex one.

One word of caution about reading a pencil; insert it in the reader in a positive manner so that the spring-loaded activation switch does not push it out again. After pencil is read, remove it and reinsert it: reading will be different the second time, and the third, and so on because

the read circuit requires some charge to activate it, and each time the pencil is removed and reinserted, it discharges slightly, making each successive reading higher (a conservative feature).

Once a reading is obtained, to recharge the pencil, push charge button and vary voltage (or zero-adjust) control to place indicating hairline on zero point of scale. With the larger, more complicated chargers, once the zero voltage is set for one pencil, it will repeat for additional pencils whenever the charge button is pushed. This statement may not be true of portable chargers: each pencil may require a voltage adjustment to zero.

When a pencil is carried into a radiation area and β or γ radiation enters the chamber through its walls, air in the chamber becomes ionized (made slightly conducting), causing the charge to leak from the center electrode. The lower the charger, the higher the indication when the pencil is placed in the reader.

Alpha particles are unable to penetrate the heavy plastic chamber wall, which is internally coated with graphite to make it the outer conducting electrode. Also, this same wall blocks low-energy radiation, such as β radiation below about 1 MeV and γ radiation below 20 keV (approximate numbers); reference should be made to manufacturer's data sheets for actual limitations. These sheets usually include calibration curves, which are helpful if a dosimeter needs to be used in a "soft" radiation environment.

THE SELF-READER DOSIMETER

There is an old saying to the effect that, "There are

only two people I trust, thee and me, and I'm not so sure of thee." This state of mind is a drawback to the pencil dosimeter for people who are radiation workers. They can be told that they are not going into a situation where they will be overexposed, but they feel more comfortable knowing what their exposure is. For this reason, the device most used by professionals is the self-reader. The pencil, in most cases, is for the visitor.

The small metal-coated quartz fibers contained in a self-reader (Figure 2-5-6) operate exactly as do the gold

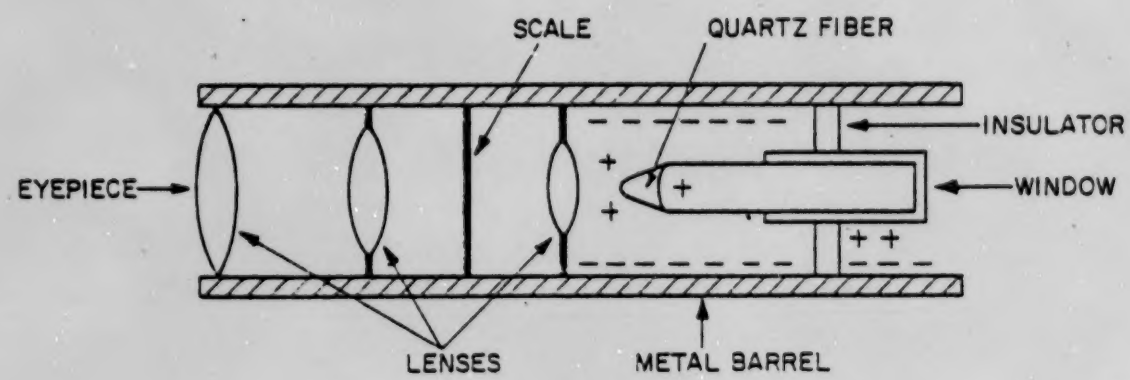


Figure 2-5-6. Self-reading dosimeter.

leaves in the electroscope. The charger is simply a battery box with a voltage adjustment, and a small light to illuminate the scale. The position of the movable fiber is adjusted with the voltage-adjust/zero-set knob until it falls on the zero mark. The scale of the self-reader is similar to that shown in Figure 2-5-5.

When zeroing, the dosimeter may read slightly upscale when removed from the charger. There are two ways to correct

for this variation. It can be "zeroed" out low so that, when removed from the charger, the fiber can move up to the zero mark; or the initial reading for the chamber can be recorded, and subtracted from the final reading.

The lens system within the self-reader superimposes a clear view of the fiber on a scale; also, it provides a clear view of both when seen through the eyepiece.

A dustcap keeps dust out of the charging connection of the pencil. (No such dustcap is required for the self-reader.) A small amount of dust on the insulator in the connection can discharge the chamber. And, when discharged, the chamber reads far in excess of the full-scale reading.

Though these dosimeters can take the normal shocks of being carried in the pocket, a good hard jolt can cause it to discharge. For this reason, when pencils are used, it is customary to carry two rather than one.

Should a self-reader be dropped, exposure can be estimated from its last reading. Also, because the self-reader is a control device rather than a legal recording device (such as the film badge or thermoluminescent dosimeter), a dropped self-reader causes no great concern.

When using an ionization type dosimeter, a phenomenon called insulator soak-in may occur. Soak-in can happen to a new chamber or one that has been out of service or remained in a discharged condition for some time. When first charged, a rather rapid discharge usually occurs because of penetration of some of the charge into the insulator, which is not leakage in the ordinary sense. The problem usually disappears if dosimeter is kept on charge for a day or so.

NEUTRON REACTIONS IN DOSIMETERS

Neutron dosimetry is, at best, difficult. For a neutron to cause ionization inside a dosimeter of the types so far described, it must react in some way. Then the ionization caused by the reaction products must be measured. Also, neutron reaction rates are highly dependent on the energy of the neutron.

There is an old rule of thumb that a thermal neutron flux of 256 neutrons/cm^2-sec will produce one mrem/hr, and a fast neutron flux of four neutrons/cm^2-sec will produce one mrem/hr. The important thing to note is that this variation is 64 to 1, a significant difference. Therefore, personal neutron dosimeters concentrate on measuring thermal neutrons because they are easier to detect. Other instruments are used to detect neutrons of higher energy. On the scale of a neutron self-reader (Figure 2-5-7) note that it reads dose equivalent of both thermal neutrons and γ radiation.

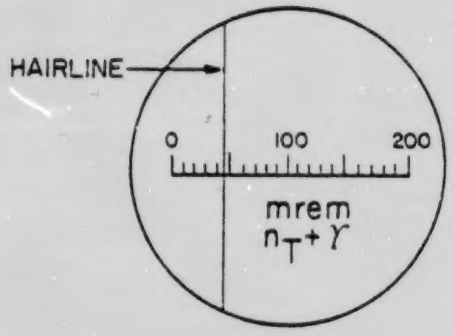


Figure 2-5-7.
Neutron self-reader scale.

EXAMPLE

Given: A survey of an area shows that the radiation field is composed of 50 mR/hr of γ radiation, 5 mrem/hr of thermal neutrons and 50 mrem/hr of fast neutrons. The dosimeter reads 6 mrem upon entry into this field and 39 mrem after leaving.

Find: Total actual exposure.

Solution: First, the net reading is 39 - 6 = 33 mrem. From the data obtained with our survey instruments, we know that the γ-to-thermal neutron radiation exposure ratio is 50 mrem/hr to 5 mrem/hr, or 10/1; therefore, the radiation breakdown would be 30 mrem of γ, 3 mrem of thermal neutrons; to that we should add 30 mrem of fast neutrons. This conclusion is valid if nothing changed while the person wearing the dosimeter was in the area.

Assuming that the radiation field did not change, then the reading of the dosimeter, together with the survey information, can be used to approximate the person's total dose at 63 mrem. Here, "approximate" means that the dosimeter will read conservatively unless we are dealing with weak γ rays, in which case the dose will be higher than indicated, and the manufacturer's calibration curves will have to be consulted to find the correct dose.

To keep a dosimeter from becoming internally contaminated, products of neutron reaction cannot be themselves

radioactive. The prime reaction that provides particles that can cause significant ionization and that can guarantee noncontamination is the boron reaction or, more specifically, the neutron reaction with the isotope ^{10}B:

$$^{10}_{5}B + ^{1}_{0}n \rightarrow ^{7}_{3}Li + ^{4}_{2}\alpha + 2.35 \text{ MeV}.$$

Equation 2-5-1
(Equation = Eq.)

The 2.35 MeV appears as kinetic energy of the ^{7}Li particle and the α particle. Neither of these particles, which are responsible for ionization in the tube, is radioactive. Boron is usually coated onto the inside of the chamber's sensitive volume in a sufficient amount to last for many years or, for practical purposes, "forever."

Personal neutron dosimetry is not usually of concern, even around a reactor; however, it is good to be familiar with personal neutron dosimetry in case a situation arises where one might expect a possible neutron exposure.

Other types of dosimeters exist for special problems. If a special problem exists that cannot be handled by a standard dosimeter, or continuous surveying is impractical, check out available devices.

EXERCISES

A plant supervisor is wearing a 0-200 mR pocket dosimeter. As the supervisor entered the reactor containment area it indicated 20 mR. As the supervisor left the compartment, it indicated 150 mR. If this person was in the area for 30 minutes, what was the dose rate in mR/hr?

MATERIALS

1 encapsulated γ source of known strength and material.
1 soaked* and 1 unsoaked** pocket ion chamber (pencil) with matching charger-reader.
1 soaked and 1 unsoaked self-reading pocket ion chamber (self-reader) with small matching charger.
Cotton (small piece).
1 stopwatch, or other suitable timing device.

---

* Soaked - means that ion chamber has been kept on charge for a day or so to ensure that "insulator soak-in" has been taken into account.
** Unsoaked - a new ion chamber that has simply been charged and put into use.

CHARGING AND READING OF POCKET ION CHAMBERS

Use a self-reading dosimeter that has been in service for some time (soaked); look through it toward a light source and identify scale units and hairline. (If no hairline is visible, dosimeter needs to be charged.)

Practice inserting dosimeter into charger as follows:

1. If charger has off-on switch, switch it on; if it doesn't, activate it by inserting fully into socket.
2. Locate voltage-adjustment or zero-set knob used to adjust position of hairline when charging dosimeter.
3. After dosimeter is placed in charger, hold it down firmly and look through it. Can you read it? If not, and charger is turned on, batteries are in need of replacement or, possibly, the illumination bulb is burned out.
4. Use voltage-adjustment knob to set hairline to zero. If hairline was off scale to begin with, you may have to turn knob in both directions until it appears.
5. Remove dosimeter from reader and look through it again. Does it still read zero? If not, reinsert it into charger and adjust hairline to a position slightly to the left of (below) zero; remove it from charger and look through it again; repeat this procedure until hairline moves exactly to zero when dosimeter is removed from charger.
6. Shake dosimeter (as though it were a thermometer). DO NOT drop it or bang it on a table or desk top (it contains GLASS in the lenses and very delicate fiber). After shaking, it should read exactly as before. Does

It? If not, you have either shaken it too vigorously, or you have a defective dosimeter.

7. Obtain a new (unsoaked) dosimeter.
8. Charge it <u>once</u>.
9. Look through it. Can you see the hairline move? If you can, estimate time it will take to reach top of scale, and recharge it before it gets there.
10. If you cannot see hairline move, note reading and set it aside in a nonradioactive area for an hour or so, then read it again. Has it drifted upscale? If so, estimate the time it will take to reach full-scale, and recharge it before it gets there.
11. In either case, whether the rate of drift is fast or slow, charge it often enough to keep reading on-scale for the next 24 hours.
12. If you obtain a dosimeter which does not drift at all, try another new (unsoaked) one.
13. When both dosimeters (the soaked and the unsoaked) are fully charged and stable, set them aside for use in the following procedure.

THE POCKET (PENCIL) IONIZATION CHAMBER

The procedure to be followed here is similar to that for the self-reader, with minor differences. Those minor differences are covered in the following procedures - read carefully.

1. Use a pencil that has been in service for some time (soaked); remove dustcap; insert pencil firmly into

charger-reader.

2. Locate off-on switch on charger-reader and place it in on position. (If there is no off-on switch, the device is activated by inserting dosimeter in socket, firmly and fully.)
3. Look through eyepiece; if you see an illuminated scale, batteries and bulb are all right. If you cannot see scale, try inserting the dosimeter more firmly; if that does not help, replace batteries or bulb or both as needed.
4. Is hairline on-scale? If not, adjust voltage control to put it on-scale. Moving voltage-control or zero-adjust knob will have no effect unless you depress charge button. Note: Exact reading does not matter, as long as hairline is on-scale.
5. Note reading.
6. Remove pencil and reinsert it. Has reading changed?
7. Repeat above procedure until you have convinced yourself that reading will change every time, or reading will not change.

   If the reading changes each time, remember to take first reading when reading pencil for the record; also, remember to insert it properly in the charger.
8. Practice inserting pencil into charger a few times, to get feel of how to insert it.
9. Locate charge button.
10. If charger-reader is properly adjusted, all that is needed to zero pencil reading is to push charge button and release it.
11. If pushing charge button does not exactly zero reading, adjust voltage control (while holding charge button down) until zero reading is obtained; reading should

remain zero after button is released.

12. Practice this adjustment a few times to get feel of it; move voltage-adjustment or zero-set knob slightly, then insert dosimeter and reset zero.

    Set the dosimeter aside (it will be used in step 15).

13. Leave voltage-adjustment knob set at zero position and obtain a new (unsoaked) dosimeter.

14. Insert unsoaked dosimeter in charger-reader and zero it. Set it aside for twenty minutes or so, then read it again. Has reading changed? If so, estimate how long it will take to reach top of the scale, and recharge it before it gets there. Repeat this procedure during the next 24 hours, until dosimeter is soaked.

15. Vigorously shake soaked dosimeter (from step 12) and read it: reading should not have changed. Now, drop it from a height of about two feet onto a table or desk top, and read it again. If it is still at or near zero, drop it onto floor from a height of five or six feet. One of these two drops should discharge the dosimeter so the hairline moves completely off the high end of the scale. <u>If not, do not subject it to any more physical abuse</u>. (Note: This experiment is performed to illustrate what can and does happen to dosimeters in use. As indicated earlier, these are delicate instruments that should not be abused.)

16. Re-zero dosimeter. Take a small piece of cotton or other fibrous material and pack it into end of dosimeter that goes into charger-reader. (Pack it loosely so that you can get it out again, but make sure that it goes all the way into the socket, at least on one side of the center pin, and that cotton contacts

the insulator.) Set dosimeter aside for 30 minutes or so.

17. Remove cotton and read dosimeter. Has dosimeter discharged? If not, reinsert cotton and leave it for a longer period of time. Repeat procedure until you have convinced yourself that lint or dust can cause the dosimeter to discharge, or that lint and dust do not affect the device.
18. Re-zero both pencils and set them aside until ready for the following procedure.

EXPOSURE TESTING

1. Use an encapsulated source about which the following is known:

    Source strength.
    Source radionuclide.
    Date on which source strength was determined.

    Use procedure outlined in Module 2-1, together with Table 2-5-1, and the inverse-square law, to determine a source-to-detector geometry that will result in an exposure rate of exactly 50 mR/hr.
2. Set up pencils as shown in Figure 2-5-8, at exactly the required distance. (You may have to fabricate holders to keep them in proper position. Be sure holders do not interfere with radiation field.)
3. When pencils are in place, place source in the center of the dosimeter array; observe due precautions for handling source.

TABLE 2-5-1

GAMMA RADIATION LEVELS FOR ONE CURIE OF SOME RADIONUCLIDES*

NUCLIDE	Γ†	NUCLIDES	Γ†	NUCLIDES	Γ†
Actinium - 227	~2.2	Gold - 198	2.3	Potassium - 43	5.6
Antimony - 122	2.4	Gold - 199	~0.9	Radium -226	8.25
Antimony - 124	9.8	Hafnium - 175	~2.1	Radium - 228	~5.1
Antimony - 125	~2.7	Hafnium - 181	~3.1	Rhenium - 186	~0.2
Arsenic - 72	10.1	Indium - 114M	0.2	Rubidium - 86	0.5
Arsenic - 74	4.4	Iodine - 124	7.2	Ruthenium - 106	1.7
Arsenic - 76	2.4	Iodine - 125	~0.7	Scandium - 46	10.9
Barium - 131	~3.0	Iodine - 126	2.5	Scandium - 47	0.56
Barium - 133	~2.4	Iodine - 130	12.2	Selenium - 75	2.0
Barium - 140	12.4	Iodine - 131	2.2	Silver - 110m	14.3
Beryllium - 7	~0.3	Iodine - 132	11.8	Silver - 111	~0.2
Bromine - 82	14.6	Iridium - 192	4.8	Sodium - 22	12.0
Cadmium - 115m	~0.2	Iridium - 194	1.5	Sodium - 24	18.2
Calcium-47	5.7	Iron - 59	6.4	Strontium - 85	3.0
Carbon - 11‡	5.9	Krypton-85	~0.04	Tantalum-182	6.8
Cerium - 141	0.35	Lanthanum-140	11.3	Tellurium-121‡	3.3
Cerium - 144	~0.4	Lutecium-177	0.09	Tellurium-132	2.2
Cesium-134	8.7	Magnesium -28	15.7	Thulium-170	0.025
Cesium-137	3.3	Manganese-52	18.6	Tin-113	~1.7
Chlorine -38‡	8.8	Manganese-54	4.7	Tungsten-185	0.5
Chromium-51	0.16	Manganese-56	8.3	Tungsten-187	3.0
Cobalt-56	17.6	Mercury-197	~0.4	Uranium-234	~0.1
Cobalt-57	0.9	Mercury 203	1.3	Vanadium-48	15.6
Cobalt-58	5.5	Molybdenum-99	~1.3	Xenon-133	0.1
Cobalt-60	13.2	Neodymium-147	0.3	Ytterbium-175	0.4
Copper-64	1.2	Nickel-65	~3.1	Yttrium-88	14.1
Europium-152	5.8	Niobium-95	4.2	Yttrium-91	0.01
Europium-154	~6.2	Osmium-191	~0.6	Zinc-65	2.7
Europium-155	~0.3	Palladium-109	0.03	Zirconium-95	4.1
Gallium-67	~1.1	Platinum-197	~0.5		
Gallium-72	11.6	Potassium-42	1.4		

* Jaeger, R. G., et al., Engineering Compendium on Radiation Shielding, Vol. 1. (New York: Springer-Verlag, 1968), pp. 21-38.

† Γ = R-cm²/hr-mCi or Γ/10 = R/hr at 1 m/Ci

‡ A Manual of Radioactivity Procedures (National Bureau of Standards Handbook No. 80 (Washington, D.C.: Dept. of Docs., U.S. Government Printing Office, Nov. 1961)), Appendix A, pp 137-140.

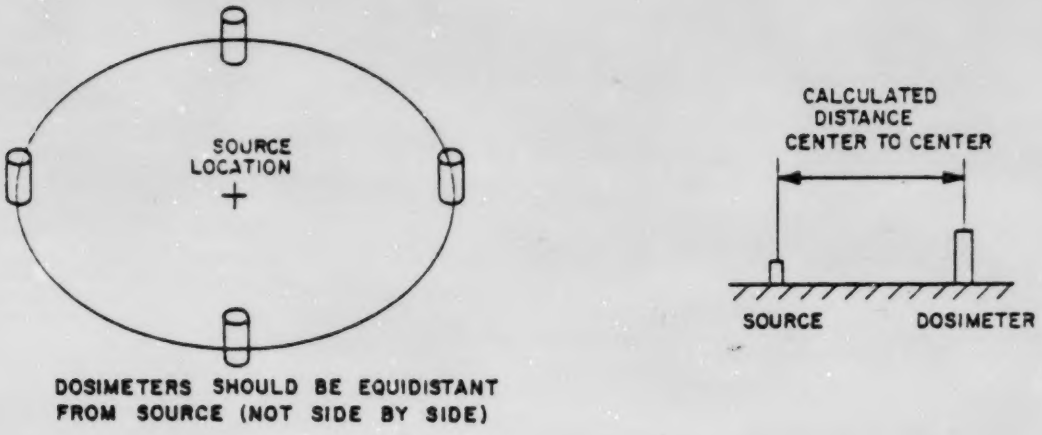


Figure 2-5-8. Geometry for exposure procedure.

4. Start exposure timing as soon as source is in place. (Since radiation protection means radiation avoidance, place appropriate warning signs and leave the immediate area.)
5. Allow dosimeters to remain in this field until they have had time to read midscale (about two hours). This procedure can be speeded up by exposing dosimeters in a higher field; however, dosimeters should never be so close to source that it is no longer a point source. Under such a condition, the inverse-square law no longer applies. If time is shortened too much, time of placing and removing source will affect reading.
6. When irradiation time has elapsed, store source.
7. Read dosimeters and record results. Are results as expected? Both dosimeters should read approximately the same, at or slightly above midscale (within 10%). If not, why not?

DATA TABLE

None.

REFERENCE MATERIALS

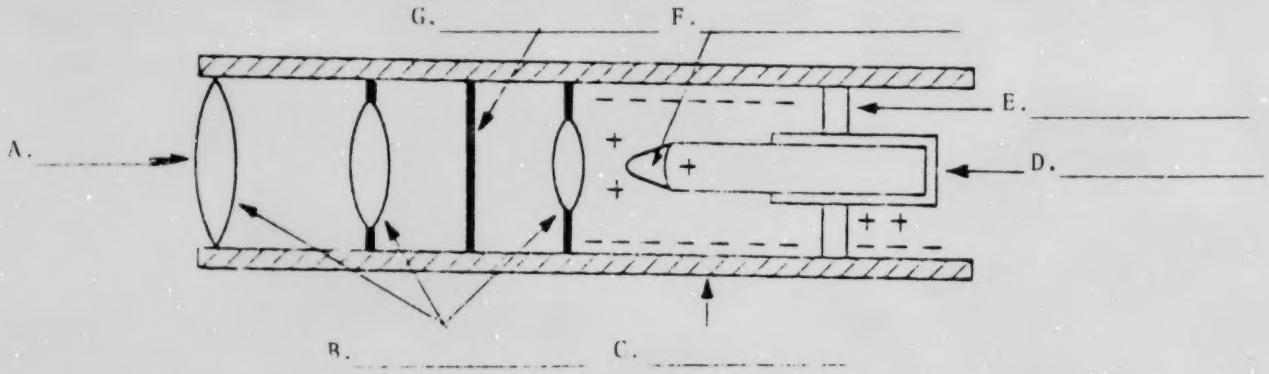
Cember, Herman. Introduction to Health Physics. New York: Pergamon Press, 1969.

U.S. Department of Health, Education and Welfare. Radiological Health Handbook. Cincinnati, OH: USDHEW, 1970.

TEST

Read each question carefully. Be sure your responses give the specific information each question requires. Your answers should be brief. If you need more space, however, write on the backs of the test.

1. Label the lettered parts of the dosimeter to which the arrows point.



2. Is the dosimeter in Question 1 self-reading; or does it require a charger reader?

3. What types of radiation does the pocket dosimeter detect? Why can't the dosimeter measure alpha radiation?

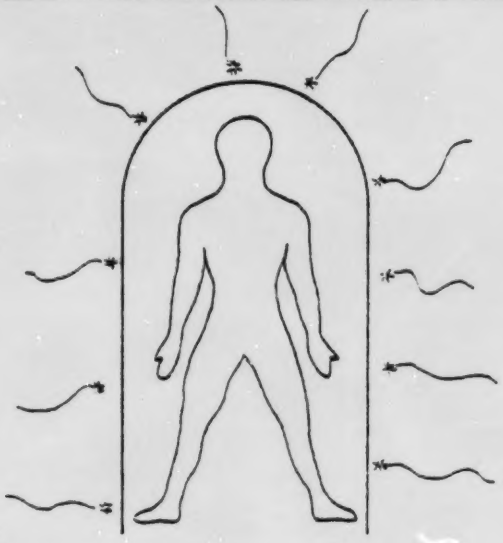
4. What is one way you can correct the variation if, following zeroing, a dosimeter reads slightly upscale when removed from the charger?

5. What does "insulator soak in" mean? What usually cause soak-in?

6. Calculate total exposure, given the following information: The survey of an area shows that the radiation field is composed of 60 mR/hr. of gamma (γ) radiation; 10 mrem/hr. of thermal neutrons and 60 mrem/hr. of fast neutrons. The pocket dosimeter reads 5 mrem upon entering the radiation field and 35 mrem after leaving the field.

NUCLEAR TECHNOLOGY

MODULE 2-6 PERSONNEL MONITORING DEVICES - CHEMICAL TYPES



RADIATION PROTECTION I

COURSE 2

COURSE MODULES

COURSE 2 RADIATION PROTECTION I

MODULE 2-1 Radiation Exposure and Absorbed Dose Values

MODULE 2-2 Biological Effects of Radiation

MODULE 2-3 Radiation Protection Standards

MODULE 2-4 Internal Dose Calculations

MODULE 2-5 Personnel Monitoring Devices, Ionization Types

MODULE 2-6 Personnel Monitoring Devices, Chemical Types

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INTRODUCTION

Deposition of energy by an interaction of ionizing radiation with atoms (or molecules composing the structure of many materials) often results in a chemical change. In the case of many sensitive chemical compounds, these changes are sufficiently visible to permit their use as sensitive indicators of the presence of radiation through color changes, changes in transparency or changes in reactivity with other chemicals. Further, a graduation of chemical changes can be used as a measure of the amount of radiation energy deposited and interpreted in terms of units of radiation exposure dose. Exposure dose can be translated into individual radiation dose in rad or rem units; then, with a knowledge of the type and energy of radiation, Roentgen exposure units can be converted to appropriate rem units, also.

In general, these sensitive substances can be characterized into three major groupings:

Chemical dosimeters.
Film dosimeters.
Solid-state dosimeters.

MODULE PREREQUISITES

The student should be familiar with the contents of the following modules:

Module 2-1, "Radiation Exposure Units."

Module 2-3, "Radiation Protection Standards."

Module 2-5, "Personnel Monitoring Devices - Ionization Types."

Module 1-3, "Atomic Structure and Radioactivity."

Module 1-6, "Basic Properties of Gamma Radiation."

Also, the student should understand and apply appropriate safety precautions for dealing with radioactive sources.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Describe five methods of detecting ionizing radiation (including neutrons), present basic operational features of each method, and compare these methods in terms of accuracy, speed and ease of employment.
2. Perform the following tasks relating to personnel monitoring devices:
   - Describe a chemical monitoring system used to detect high-level radiation.
   - Explain how gamma or x-ray photons, beta particles and neutrons form latent images in photographic emulsions.

List three characteristics of a photoluminescent dosimeter and explain its operating mechanism.

List five characteristics of a thermoluminescent dosimeter and explain its operating mechanism.

Explain the principle of electrical conductivity dosimeters.

Explain the principle of a photomultiplier tube; draw schematic of photomultiplier tube, depicting multiplying process.

3. Explain purpose and practical significance of the following portions of a film badge:

   Metallic shields.

   Indium foils.

   Open-window portion of film badge.

   Cadmium-covered gold element.

4. Briefly describe four factors affecting film badge development (include procedure used to ensure accurate dose identifications).

5. List five advantages and three disadvantages of film badges as radiation monitoring devices.

6. Given a radiation monitoring situation, select the most practical chemical type personnel monitoring device.

CHEMICAL DOSIMETERS

A solution may undergo chemical reaction caused by deposition of energy from radiation passing through the solution. The reaction may be observable by either a change in color of the solution (or by a change in acidity or alkalinity of the solution, which can also result in a change in color if a suitable indicator is added to the solution).

In most liquid dosimeters the effect produced is a specific radiation effect in water. The primary reaction caused by radioactive irradiation of water (H_2O) is the formation of free radicals, H and OH. These radicals may react to form hydrogen (H_2) and hydrogen peroxide (H_2O_2), or they may recombine to form water.

Note: Hydrogen may act as a reducing agent on certain chemicals, and hydrogen peroxide may act as an oxidizing agent on other chemicals (Figure 2-6-1).

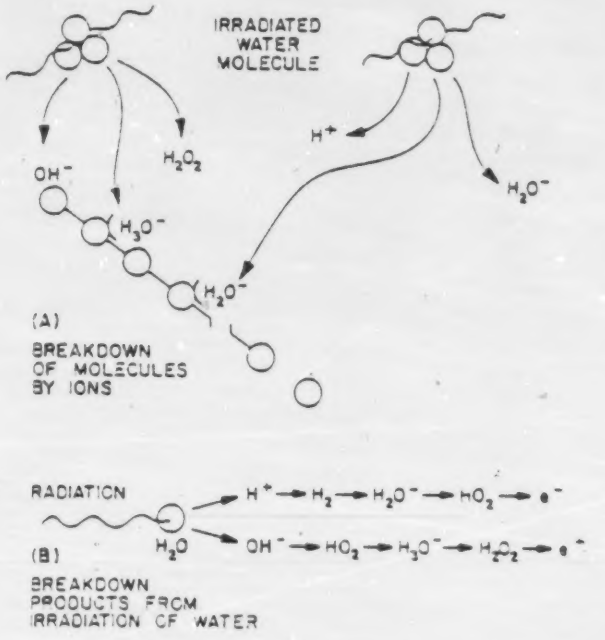

Figure 2-6-1. Reaction caused by radioactive irradiation of water.

One common chemical dosimeter consists of a mixture of water and a chlorinated hydrocarbon, such as chloroform, trichlorethylene or tetrachloroethylene. In such solutions, hydrochloric acid (HCl) is produced by the combination of H with atoms of chlorine. The formation of hydrochloric acid is then indicated by means of organic dyes that change color as the pH of a solution is lowered (increased hydrogen ion concentration).

Another chemical dosimeter used in radiation detection contains a solution of ferrous sulfate ($FeSO_4$) and sodium chloride (NaCl) dissolved in sulfuric acid (H_2SO_4). The ferrous ion (Fe^{++}) is converted to a ferric (Fe^{+++}) ion by gamma or x radiation, producing a visible change in color or transparency of solution.

Chemical dosimeters are usually prepared by sealing their reagents into quartz glass vials, plastic films, or in blocks of waxy gelatin. These dosimeters are so insensitive that about 25 R of x or gamma radiation is frequently required to effect an observable change. Consequently, their use is largely restricted to nuclear reactor sites, civil defense stations and various research situations.

FILM DOSIMETERS

Film badges are most widely used of all personnel radiation detection devices. The principle involved is essentially the same as that involved in taking pictures with an ordinary camera, except the source of exposing energy is radiation in the x- and gamma-ray region of the electromag-

netic spectrum (not the typical optical region).

Radiation energy is deposited in the film emulsion, resulting in a darkening (exposure) of the film. A gelatin emulsion of a silver halide (usually silver bromide, AgBr) is dispersed in a thin film on a sheet of cellulose.

Deposition of energy through ionization by either radioactive particles or photons results in a collection of free electrons at interfaces of the silver bromide crystals. Here "sensitive regions" occur due to the presence of impurities such as silver sulfide, Ag_2S.

Ultimately, on development of the film, these clusters show up as dark patterns (clusters of elemental silver) on the film. These dark patterns are then interpreted by densitometers as being caused by a specific amount of absorbed dose. A densitometer is simply an instrument that shines a light beam through the developed film and measures the amount of light that passes through.

By using standard films that have received known amounts of radiation of comparable energy and comparing these with the film dosimeters, a reading may be obtained that can be interpreted in terms of exposure dose. This reading is based on the relationship of I_o to I, where I_o is the original intensity of the beam of light before it passes through the film, and I is the intensity of the same beam of light after it has passed through the film. Thus, transmission $T = I/I_o$ and optical density (darkening of film by irradiation) $D = \log_{10} 1/T$. Film density, then, is interpreted directly in terms of Roentgen units of exposure.

To understand the mechanisms of film-badge dosimetry, materials composing the film must be examined. Also, the theory of image formation must be postulated and the limitations of the system defined.

EMULSION

A variety of film emulsions exist. One feature, common to all, is the gelatin-base silver halide that covers the film (cellulose) or glass plate. Emulsion thickness varies in different types of film, from a few microns to several hundred microns. The common emulsion-thickness range for films used as radiation detectors is 10 to 25 microns. This thickness corresponds to an areal density of from 2 to 5 mg/cm^2. Usually, a thin protective coating of gelatin, sometimes referred to as a T-coat (about 0.5 microns thick), covers the emulsion.

Grain size of the silver bromide is important in determining film sensitivity. (Sensitivity is a measure of how much radiation is needed to produce a given level of exposure.) Impurities, as well as developing techniques, may modify sensitivity.

The constituents of emulsion may be varied, depending on the type of radiation and the levels of radiation monitored. For example, in recording the tracks of heavy particles, grain concentration is adjusted to about three times that used in ordinary photography film. In addition, certain sensitizers may be added, such as boron, which greatly increases film sensitivity to thermal neutrons.

LIMITATIONS

Energy Dependence

Sensitivity of a particular film to x-rays is a function of their effective energy (Figure 2-6-2). The initial gradual rise in sensitivity (as the energy decreases) is primarily caused by a photoelectric interaction with silver at the

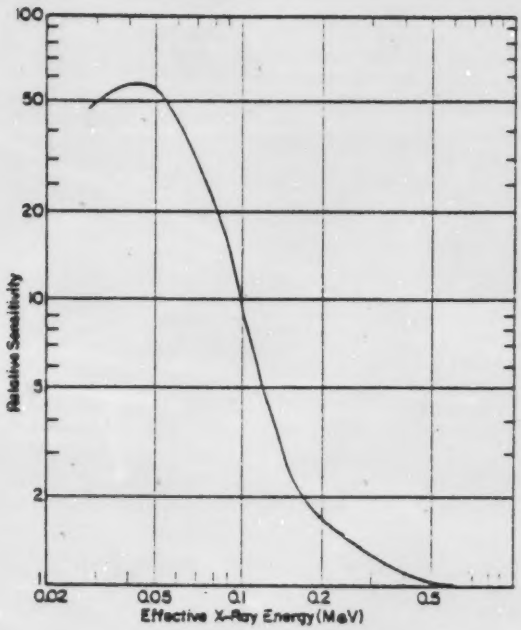

Figure 2-6-2. Sensitivity of a particular film to x-rays as a function of x-ray energy.

lower energies. The decrease in sensitivity below 50 keV is caused by absorption within the outer film covering, preventing radiation from interacting with the underlying sensitive emulsion. Note: Sensitivity tends to decrease slowly as energy of photons is increased about 1 MeV.

If film is used to measure dose from x-ray photons of known energy and is calibrated for this energy range, measurements are reliable. On the other hand, if a wide range of energies are to be encountered, energy dependence should be reduced so that film response is essentially flat. This adjustment can be accomplished (over the spectrum) from about 50 keV to 3 MeV. (Energies outside of these limits should be measured with other techniques because

results with film may be misleading and inaccurate.)

Several methods have been used to reduce energy dependence. A lower silver bromide content is of some value, but may affect resolution as well as sensitivity. Organic scintillators have been used with some success. Adding a scintillator affects the resultant sensitivity curve (Figure 2-6-3). Ultraviolet fluorescence from the scintillator effectively increases sensitivity and flattens energy response.

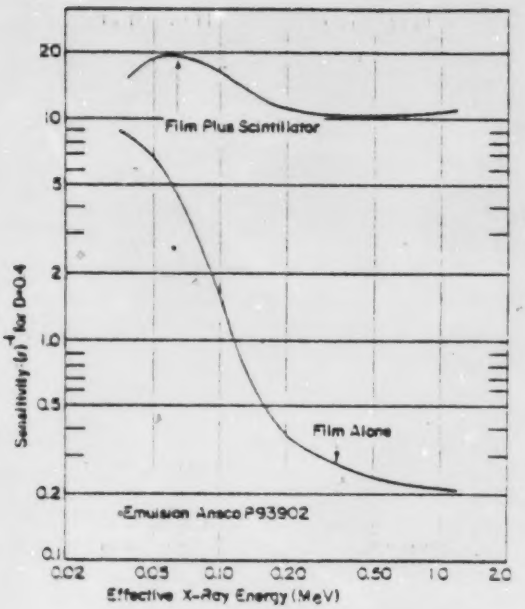

Figure 2-6-3. Sensitivity of film versus x-ray energy, for film with and without scintillator.

The most common approach to the problem is a system of filters in front of the film. These filters, calibrated

to yield information on the energy of unknown photons, as well as total exposure, permit more complete evaluation of an exposure dose.

Angular Dependence

Actually, the dose to a material (tissue or emulsion) is dependent on angle of incidence, because dose is computed first for an infinitesimal slab, then integrated over the entire volume. Film darkening is also dependent on angle of incidence. Table 2-6-1 illustrates this dependence.

TABLE 2-6-1

RELATIVE FILM-BADGE SENSITIVITY FOR GAMMA RAYS INCIDENT AT VARIOUS ANGLES

Angle of Incidence	Sensitivity Gamma-Ray Energy		
	0.11 MeV	0.20 MeV	1.20 MeV
0° perpendicular, indicence	1.00	1.00	1.00
22.5°	0.87	0.92	0.97
45°	0.46	0.73	0.91
90° orthogonal incidence = glancing	0.16	0.41	0.84

Rate of Exposure

Evidence indicates that photographic effect is independent of the rate at which exposure is produced. There is little knowledge of the effect of microsecond exposures to radiation; but, due to the possibility of repetitive exposure

(one grain hit by more than one particle or photon), it is probably of some significance.

After an exposure of several days, some film fading occurs. This fading is influenced by both temperature and humidity; even so, by selecting films of different sensitivities, effective film dosimeters for wide dose ranges can be devised. For example, DuPont 502 film can be used from 100 mR up to 10 R, and Eastman 548-0 double coat is suitable from 500 R up to 10,000 R.

Developing Techniques

Even slight deviations in developing time, type, quality and temperature of developer may affect density in film dosimetry; therefore, the usual procedure is to develop a set of known standard films with each batch of unknown films.

Material Surrounding Film Packet

Normally, tissue-equivalent or air-equivalent material is used as the film holder. This material - usually lucite, or bakelite plastic with an open window in the holder - permits detection of beta particles. Thickness of the plastic must be sufficient to permit electrical neutrality to be achieved for the particular energy of the incoming particles or photons. Obviously, in a mixed radiation field, this ideal is difficult (often impossible) to achieve.

THEORY OF LATENT IMAGE FORMATION

Radiation loses energy in interacting with emulsions by setting electrons free or raising the energy of one or more electrons in the crystalling silver bromide. Usually, more than one ionization event is required. Electrons mi-

grate in the crystal, and eventually become trapped in "sensitive regions." These regions may consist of impurities in the crystal lattice (such as silver sulfide). The electrostatic potential set up around the centers results in an accumulation of silver ions, which can migrate in the crystal (Figure 2-6-4). These ions collect in the sensitive region due to their attraction for negative electrons. Ultimately, the ions become neutralized to form elemental silver atoms.

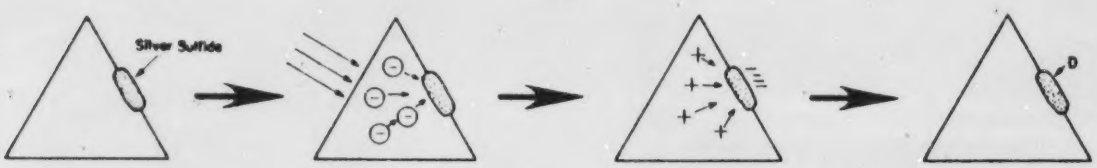

Figure 2-6-4. Electron migration of photosensitive crystal.

The silver atoms constitute the latent (undeveloped) image of the emulsion and act as a catalyst during the development process to completely convert the transparent, activated silver halide grain to visible elemental silver. The development process is merely an amplification (of about 10^{12}) of what has already transpired. The probability of a latent image being established increases with the amount of energy absorbed by the emulsion from the radiation.

NEUTRON FILM DOSIMETRY

Measurement of thermal neutron dosage may be accomplished by comparing film densities produced by cadmium filters with

those produced by brass filters, or cadmium filters with gold filters. These filters are designed to attenuate gamma radiation by a similar amount; however, because of the (n-gamma) reaction induced in cadmium by the thermal neutrons, the film density behind brass filters (when thermal neutrons are present) is lower than behind the cadmium filter. If a wide spectrum of gamma energies are present, a modification of this approach (utilizing cadmium-gold filters) can be used. As an alternative method of measuring neutron dosage, particle-track plates may be used.

Because of the low sensitivity of photographic film to fast neutrons, fast-neutron dosimetry is accomplished by track counting under a microscope, rather than by determining optical densities of films. The predominant interaction between fast neutrons and photographic emulsion is the elastic collision with hydrogen atoms in the emulsion. The recoiling protons (nuclei of the hydrogen atoms) have, on the average, one-half the energy of incident neutrons. These charged particles (protons) interact with photographic grains. The minimum energy of incident neutrons must be 0.5 MeV or higher to make track counting reasonably accurate.

Thermal-neutron dose also can be determined by using the captured alpha reaction from the interaction of neutrons with boron. In this method, a portion of the surface of the dosimetry film is covered with a boron-loaded, silver-activated, zinc-sulfide screen. Alpha particles (produced in the boron during the neutron capture process) cause scintillations (in the zinc sulfide) that are recorded by the photographic film.

This method, just as most other photographic dosimetry methods involving scintillators, has the advantage of high sensitivity. But, it has the disadvantage of exhibiting

a pronounced rate of sensitivity dependence, introduced through exposure of the emulsion to visible or near visible scintillator light.

TYPES OF FILM BADGES

Four major types of film badges are worn by personnel in a reactor area.

Lapel Badge

A lapel badge, sensitive to beta, gamma and x radiation, consists of a paper-covered packet containing one piece of high-sensitivity film and one piece of low-sensitivity film. The packet is enclosed in a plastic holder into which are inserted two metal filters. These filters adjoin the film packets on either side so that the front and back filters are the same. After exposure, the films are developed along with pre-exposed calibration standards, and density in the film behind each filter is measured and compared with the calibration standards. These independent measurements permit a crude evaluation of exposure as a function of energy and type of radiation.

Construction of this badge (Figure 2-6-5) is fairly

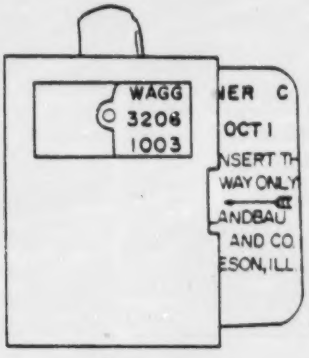

Figure 2-6-5. Components of a lapel badge.

simple. When inserting a fresh film packet, the packet worn during the previous period is ejected from the left side of the holder. Once inserted, the packet should not be moved within the holder until a new packet is inserted.

Ring Badge

A smaller version of the lapel badge can be worn as a ring (Figure 2-6-2). Its usefulness is restricted to situations in which the hands are most likely to receive the radiation exposure to the exclusion of the rest of the body.

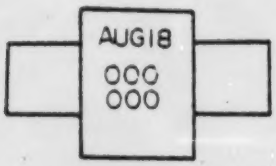

Figure 2-6-6.
Ring badge.

Other Type Badges

Some lapel badges are sensitive to beta, gamma, x and fast-neutron radiation. These badges are the same as the previously described lapel badge, except that they incorporate an additional film packet (sensitive to fast neutrons) and an additional metal filter.

Another type of lapel badge is sensitive to beta, gamma and x radiation and fast and thermal neutrons. This badge is the same as that preceding, except that a special technique, using the neutron-boron interaction, is employed to evaluate the contribution of thermal neutrons to film density.

BADGE ACCURACY

Reasonable accuracy of film badge dosimetry (± 20%) can be obtained by adhering to the following guidelines:

Employ well-trained, experienced film-badge evaluation personnel.

Use high-quality badges with optimum multifilter designs.

Be sure exposure to radiation beam is as near perpendicular as possible.

Use single film for 0.05-to-20 R dose range; use double film for 0.05-to-1,000 R dose range.

Limit use of film badges to energies ranging from 25 keV to 2 MeV.

Store control standards and film badges under identical conditions.

Process standard calibration films simultaneously with film badges, and under same conditions.

Film badges have advantages and disadvantages:

Advantages

Estimates exposure.

Measures x-rays, gamma, beta and neutrons.

Can be worn on trunk of body.

Small and lightweight.

Provides permanent, legal record of exposure.

Relatively inexpensive.

Checks work habits of employees.

Disadvantages

Large energy dependence.

Errors easily introduced in film development

and in film storage.

Neutron determinations are difficult and tedious.

SOLID-STATE DOSIMETERS

Solid-state dosimetry is based on visible or electrical change in certain solids produced by ionizing radiation. Three principal solid-state systems are currently in use:

Photoluminescence.

Thermoluminescence.

Electrical conductivity.

Solid-state dosimeters generally operate in the following manner. Ionizing radiation interacts with atoms in certain solids, freeing the electrons (orbitally). In these solids, the electrons migrate through a crystalline structure and eventually are trapped at sensitive centers (centers created by the presence of certain impurities or defects in the crystalline structure). These higher-energy-level electrons remain trapped for a considerable length of time, or until released by deposition of another form of energy. This other form of energy may be either ultraviolet or thermal.

Release of a trapped electron by heat or ultraviolet light allows it to fall back to its normal unexcited (ground) state. This release gives off energy in the form of a tiny flash of light emanating from the phosphor within the crystalline structure. By using a photomultiplier tube (Figure 2-6-7), these flashes of light are converted to an electrical signal and amplified hundreds-to-thousands of times to a

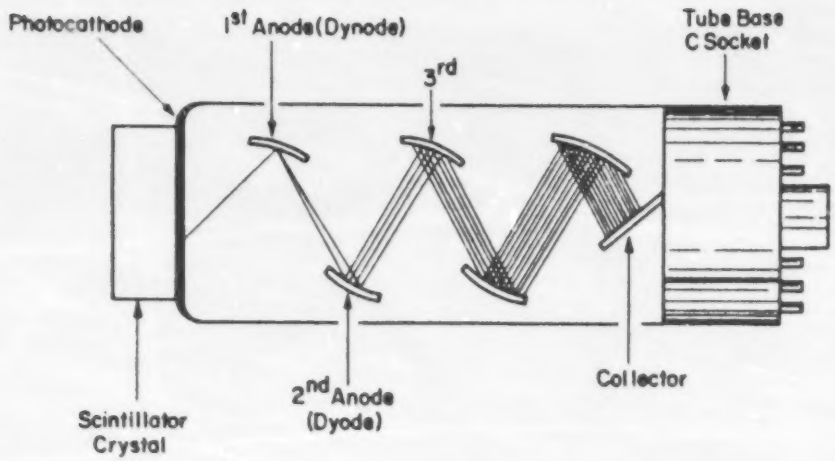

Figure 2-6-7. Photomultiplier tube.

level where they form a sufficiently intense electric pulse that can be recorded (counted). An electron freed by interaction of the light with the tube phosphor is attracted to the first anode (dynode). As it strikes this anode, two additional electrons are dislodged. They, in turn, are accelerated to the second dynode. As they strike the second dynode, each may produce two additional electrons, which results in four electrons accelerating to the third dynode. This multiplication process proceeds through a series of geometrically-situated dynodes until the last dynode produces hundreds of electrons that are collected to form a detectable current pulse.

PHOTOLUMINESCENCE

Silver-activated phosphate glass is frequently used as the solid with which ionizing radiation interacts. After

irradiation, the glass emits an orange fluorescent light continuously while exposed to ultraviolet light. The intensity of the orange light is detected by a photomultiplier tube, and a measure of the absorbed dose of radiation is obtained. Changes in the glass by radiation are both permanent (it is stored) and additive (integrated). Radiation energy deposited in the silver-activated glass rods excites the electrons of the phosphor atoms to higher energy levels. When a glass rod is exposed to ultraviolet light inside a "reader" instrument, the atoms de-excite by emitting flashes of orange light.

Glass dosimeters are manufactured in many sizes and shapes. Because they can be made as small as a ball-point tip, they are widely used in biological research work where they may be inserted directly into living tissue.

The major disadvantage of this type of dosimeter is its inability to accurately measure doses below 100 mR. It also has a tendency to fade with time; that is, the atoms may de-excite slowly over a period of time before insertion into the readout instrument. Consequently, accuracy in the total absorbed dose determination can be lost.

THERMOLUMINESCENCE

Thermoluminescence-type dosimeters may use micro crystals of calcium fluoride (CaF_2), contained in glass and mounted in a heating element in an evacuated plastic envelope. Or they may be in the form of lithium fluoride (LiF) loaded glass rods, which may be reused hundreds of times. A recent modification in the geometry of these dosimeters allows the phosphors to be encapsulated in discs of teflon in various sizes and shapes.

Incident radiation, which reacts with the acute phosphor, traps electrons at high energy levels (excited states). This form of radiation-induced stored energy is then released by heating the rods or discs. As the electrons return to their former lower-energy states, they release energy in the form of light. This light is proportional to the absorbed radiation dose, and is converted to a reading by photomulitplier tubes and various readout devices, as in the case of photoluminescence.

The CaF_2 dosimeters have a useful range of 1 mR to 3×10^5 R. Dosimeters that use lithium fluoride as a phosphor have a 10 mR to 10^3 R range; however, these dosimeters fade at a rate of only 5% a year when stored at room temperature.

After a period of exposure, a personal thermoluminescence dosimeter (Figure 2-6-8) may be plugged into an instrument that provides electrical current for the heating coil. Readout is made directly by the dosimeter heating

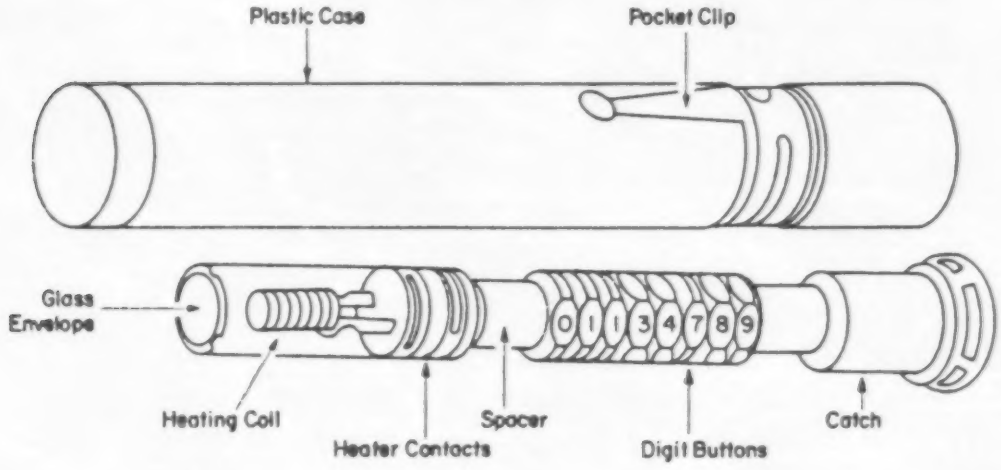

Figure 2-6-8. Personal thermoluminescent dosimeter.

instrument. Numerical buttons on the instrument may be used to set serial numbers assigned to each occupationally-exposed worker.

In a thermoluminescence dosimeter readout system (Figure 2-6-9), the calcium or lithium fluoride discs, or small glass rods, are placed on a planchet and inserted into the readout chamber through a drawer. Filters screen out any

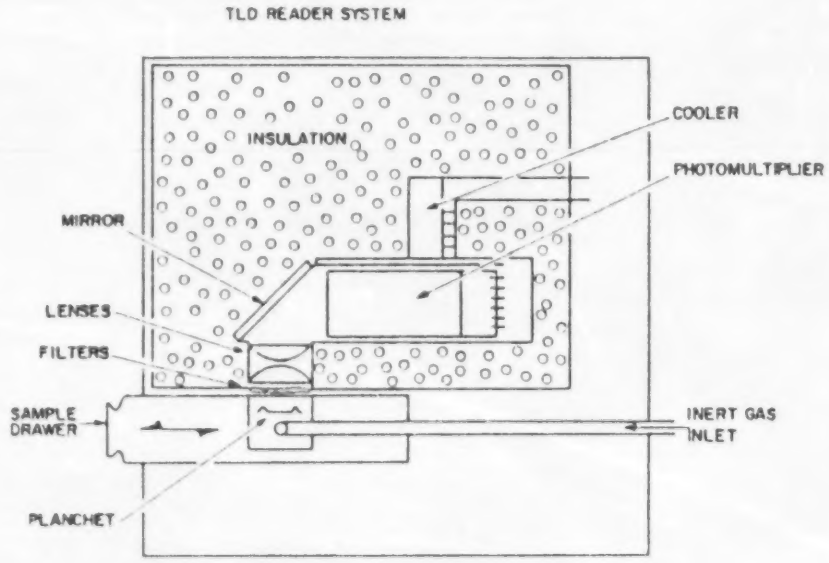

Figure 2-6-9. Thermoluminescent readout system.

light other than the fluorescent light directly produced by the irradiation process. Lenses and mirrors convey the emitted light photons to the photomultiplier tube and, subsequently, a readout is obtained on a meter or counter. Inert gas (in the chamber) stabilizes heating-coil tempera-

ture to prevent hot pockets from exceeding the desired temperature range.

Advantages of a solid-state thermoluminescent dosimeter, compared with film, include:

Higher accuracy: ± 3% in 50 mR to 10^3 R range.

Higher sensitivity: 10 to 100 mR.

More uniform energy response: almost completely energy independent in range from 10 keV to several MeV.

Greater dynamic range: can measure absorbed doses of 10 mR up to 10^6 R.

Minimal directional dependence.

Minimal fading.

Storage stability.

Simplified evaluation: direct readout of the absorbed dose is easily obtained.

Phosphor-teflon thermoluminescent dosimeters combine the desirable physical characteristics and chemical inertness of teflon with the dosimetry properties characteristic of phosphors used in thermoluminescence. Their cost, in many cases, is sufficiently low to allow discard after one use.

Lithium fluoride (LiF) teflon micro-rods are often employed as emergency dosimeters in personnel badges because of their small size (1 mm x 6 mm), high sensitivity and low cost. Lithium teflon discs are used in personnel badges, finger badges and hand dosimeters. Lithium fluoride teflon disc dosimeters also have been incorporated in laminated plastic identification cards for long-term and emergency dose measurements.

Lithium fluoride teflon dosimeters are extremely rugged, unaffected by such environmental factors as high humidity,

temperature fluctuations and mechanical shock. The front covers of these badges contain wearer's name and identification number. (This number may be coded in binary form by a series of notches for automatic printout.)

Precautions

It is important to remember that the mass of the phosphor used in thermoluminescence must be known to evaluate the absorbed dose of radiation correctly. It is obvious that the greater the mass, the more material there is with which the ionizing radiation can interact and, subsequently, the greater amount of energy that can be transferred to the phosphor.

Another crucial factor is the mg/cm^2 factor (mass of phosphor per square centimeter of surface of disc or glass rod). If phosphor mass is too great, flashes of light emitted by atoms in the bottom layers of the phosphor can penetrate to the surface and be detected by the photomultiplier tube.

Advantages of a radiophotoluminescent dosimeter (compared with a thermoluminescent dosimeter) include:

Low sensitivity - 10 R or greater.
Glass coated with silver phosphate.
Mechanism: radiation causes glass to become luminescent in proportion to total exposure.
Small size: good for biological work.

ELECTRICAL-CONDUCTION DOSIMETERS

The electrical conductivity of cadmium sulfide crystals increases according to rate of exposure (of crystals) to

neutron radiation. This principle is used in cadmium-neutron dosimeters. The change in electrical conductivity after neutron irradiation is interpreted in terms of absorbed dose. The change is brought about by the neutron interaction with the cadmium atoms.

In a somewhat similar manner, neutron interaction with indium foil activates the foil. This neutron activation of indium atoms results in emission of beta particles during their decay process. The beta particles may then be counted with a proportional counting instrument and the count interpreted as reflecting the absorbed dose of neutrons.

Indium foil may be used as a film-badge filter, with resulting production of beta particles producing a measurable effect on the film (caused by ionization produced in film emulsion by beta particles).

EXERCISES

None.

MATERIALS

1 single-film badge.
1 double-film badge.
1 neutron-film badge.
1 radiation source of known intensity.
1 densitometer.
Lithium fluoride discs or rods.
1 thermoluminescent dosimeter reader.

PROCEDURES

FILM BADGE DOSIMETRY

1. Disassemble a single-film packet; identify each component.
2. Repeat this procedure with a double-film packet.
3. Repeat this procedure with a neutron-film packet.
4. Explain differences between single-, double- and neutron-film packets.
5. Expose a series of film badges at set distances from a radiation source of known radiation intensity; calculate amount of radiation exposure at each set distance.
6. Develop film badges and read density of film (use densitometer). On graph paper, plot density factors versus calculated radiation exposure doses.

THERMOLUMINESCENT DOSIMETRY

1. Draw schematic of a TLD reader system; label all major components.
2. Expose a series of lithium fluoride discs or rods to a radiation source of known radiation intensity.
3. Calculate radiation exposure dose at each set distance.
4. Place discs or rods in thermoluminescent dosimeter reader; read and record each exposure dose.
5. Compare recorded exposure doses with calculated doses.
6. Repeat procedure (1 through 5); use the same lithium fluoride dosimeters. Is there any change in dosimeter effectiveness?

DATA TABLE

None.

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TEST

Read each question carefully. Be sure to provide the specific information each question requires. Make your answers brief. Use the backs of the test if you need more space.

1. Briefly describe the chemical reaction that takes place when a chemical dosimeter detects radiation.

2. Describe the appearance of developed film from a film dosimeter that has encountered radiation.

3. What is a densitometer? How is the densitometer used in film dosimetry?

4. List one way to reduce energy dependence in film dosimetry?

5. How do each of the following factors affect the outcome or results of film dosimetry: angular dependence, developing techniques, and material surrounding the film packet?

Angle of Incidence: _____

_____

Developing techniques: _____

_____

Material Around Film Packet: _____

_____

6. Briefly describe one method of measuring thermal neutrons in neutron film dosimetry.

_____

_____

_____

_____

7. For what energy range would film badge dosimeters be used?

_____

8. List two types of film badges.

a. _____

b. _____

9. What is the purpose of the photomultiplier tube in solid state dosimetry?

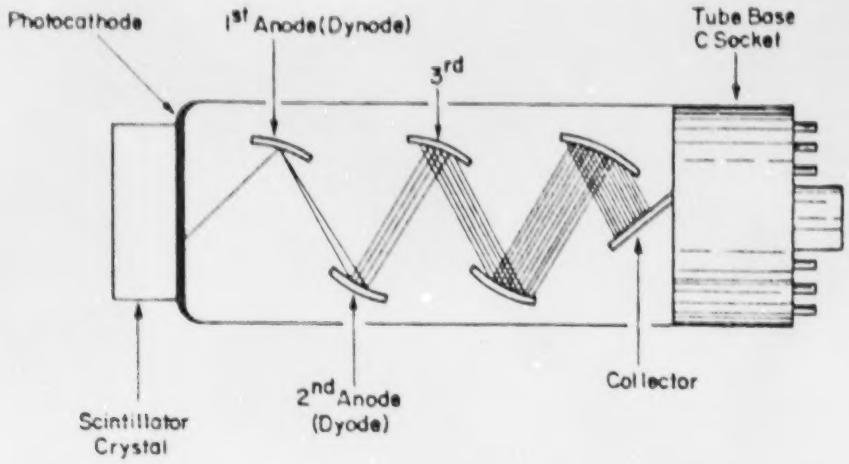
_____

_____

_____

_____

10. Using the schematic drawing of the photomultiplier tube below, explain, in words electron activity from the first anode to the collector.



Explanation: _____

_____

_____

_____

11. List three common types of solid state dosimeters.

a. _____

b. _____

c. _____

12. What is one major disadvantage of the photoluminescent dosimeter?

_____

_____

13. Compare the photoluminescent dosimeter and the film dosimeter as to the following qualities: accuracy, sensitivity, dynamic range, and uniformity of energy response.

Accuracy: _____

Sensitivity: _____

Dynamic Range: _____

Uniformity of Response: _____

14. Briefly explain how latent images are formed on films used in film badges.

15. Give the purpose of the following parts of the film badge: Cadmium covered gold element; indium foil; open window portion of badge

a. Cadmium - covered gold element:

b. Indium foil:

c. Open Window: